NUCLEAR DATA AND MEASUREMENTS SERIES

ANL/NDM-45

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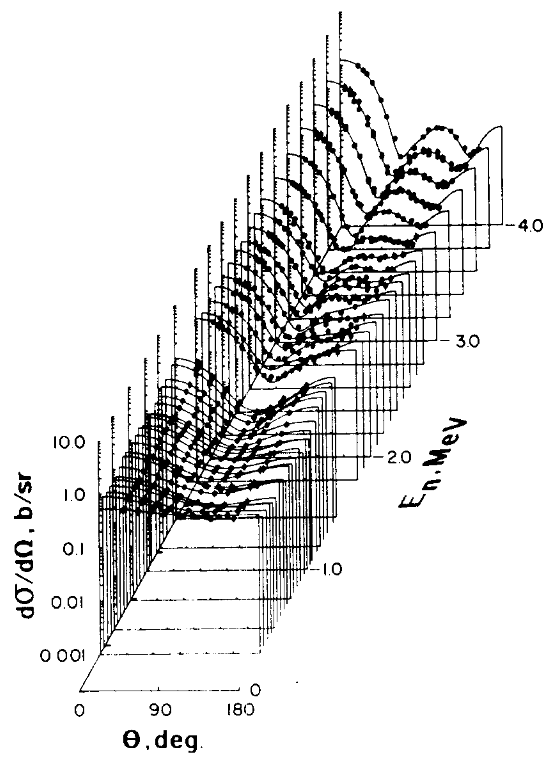

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NUCLEAR DATA AND MEASUREMENTS SERIES

The Nuclear Data and Measurements Series presents results of studies in the field of microscopic nuclear data. The primary objective is the dissemination of information in the comprehensive form required for nuclear technology applications. This Series is devoted to: a) measured microscopic nuclear parameters, b) experimental techniques and facilities employed in measurements, c) the analysis, correlation and interpretation of nuclear data, and d) the evaluation of nuclear data. Contributions to this Series are reviewed to assure technical competence and, unless otherwise stated, the contents can be formally referenced. This Series does not supplant formal journal publication but it does provide the more extensive information required for technological applications (e.g., tabulated numerical data) in a timely manner.

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EVALUATION OF $^{235}U(n,f)$ BETWEEN 100 KeV and 20 MeV*

by

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ABSTRACT

The $^{235}U(n,f)$ cross section was evaluated in the energy range from 100 keV to 20 MeV. Experimental data were included up to the 1978 Harwell Conference on Neutron Physics. The evaluation methodology is discussed in detail. The shape and the normalization of the cross section are evaluated in separate steps. An extensive comparison of the evaluation result with experimental data sets is made.

The shape of the cross section obtained in a preliminary version of the present evaluation and a normalization factor extracted from data provided within the framework of this evaluation were used by the Subcommittee on Standards and Normalizations of the Cross Sections Evaluation Working Group to establish $^{235}U(n,f)$ for ENDF/B-V above 100 keV.

*This work supported by the U. S. Department of Energy.

I. INTRODUCTION

The requirement for accurate neutron nuclear data is obvious since such data is widely utilized in applications. Hence, many attempts have been made to understand and describe observations made in experiments. This need resulted in a large number of measurements of cross sections directly required in technical applications or used in other measurements, e.g. standards. Theoretical models were developed in order to explain the experimental data, and do predict subsequently nuclide behavior which might be difficult to measure. Evaluated data files were created to supply numerical values needed for applications which represented usable compromises between often contradictory experimental data and theoretical predictions. Earlier evaluations had to use approximations which were more pragmatic than exact as data were sparce in some areas and discrepant in others. However, as experimental data became more abundant and consistent, it was recognized that the general process of arriving at an evaluated data file needed more consideration.[1] Data were often unintentionally correlated due to the use of common mass scales, neutron flux detection techniques or approximations for the interpretation of the experiments. Absolute measurements were made for many nuclei for which relative cross section ratio data were available. This resulted in an overdetermined data system with more measured values than unknown quantities which required a simultaneous evaluation in order to obtain a consistent set of data.

An attempt to evaluate a consistent cross-section-data set for the use in reactor calculations was made in 1968[2], and again with improved techniques in 1970.[1] Procedures were applied which were similar to those used in the evaluation of thermal cross sections and parameters.[3] An independent but similarly motivated evaluation was also reported in 1970.[4] The present evaluation is a first step in a new evaluation of a consistent set of neutron cross sections for the standards (^{6}Li(n,α), ^{10}B(n,α), ^{197}Au(n,γ)) and for other reactions important for technical applications (^{235}U(n,f), ^{238}U(n,γ)). The present report discusses the specific evaluation of ^{235}U(n,f) and describes the general procedures used in the evaluation of specific cross sections and ratios. A later report will contain the evaluation of the other cross sections and the consistency fit. A new simultaneous evaluation seems desirable because the data base has substantially improved since 1970[5], and because improved evaluation procedures have been developed.[5,6] More recently, a variety of somewhat differently motivated considerations[7-10] have been helpful in defining the process described and used in the present evaluation.

II. OBJECTIVE EVALUATION PROCEDURES

The knowledge of the magnitude of a quantity comes directly from one or several measurements, or indirectly from a theoretical model, or from established empirical, systematized rules. The theoretical model and the empirical rules depend on other experimental data for the determination of their parameter sets. Therefore, the knowledge one has of a quantity is ultimately the result of experimentation.[11] The measuring process, for various reasons, is subject to uncertainty, thus when the measurements are repeated they will yield different values with different uncertainties. It is the major task of the evaluation process to derive a single "best value" from the available experimental values and from predicitons from theoretical and empirical models. An objective evaluation procedure is based upon these available data and avoids intuition which might be based upon experience but is probably very subjective in nature.

1. Experimental Quantities and the Evaluation Process

The energy dependence of a quantity (cross section, cross section ratio) is usually called its shape. The quantity is known in the total energy range in which the shape is known if the absolute value of the quantity (magnitude) is known at any energy in this range. This corresponds well to the experimental determination of fast neutron reaction cross sections (or ratios). A factor in the determination of the cross section, common to all energies, is the mass of the sample. In many experiments the efficiency for detecting the reaction rate is essentially independent of the neutron energy (e.g., activation cross section experiments) or varies only slightly with energy (e.g., fission cross section experiments). Thus, a major part of the uncertainty and of unknown errors is common to all energies and the cross section can be expressed as

$$\sigma(E) = a \cdot S(E) \quad ,$$

where "a" lumps together the common factors like mass, efficiency, etc., and S(E) represents the energy dependence (shape). This situation is reflected in the data base. Many of the available data sets were the result of measurements of the energy dependence of the cross section and absolute values were not obtained. In other experiments, absolute values were obtained and measurements were repeated at different energies, thus also providing information about the shape of the cross section. A third group of experiments lead to absolute values but only at one specific energy. All three types of experiments provide valuable information. Experiments which provide only shape information might be able to do so with low systematic uncertainties but the specific setup would result in a large uncertainty of the absolute counting efficiency (and thus the absolute values). Experiments which yield absolute values only at a specific energy may use different techiques which provide values with low systematic, or at least different, uncertainties and different unknown errors but are not applicable over wide energy ranges (e.g., 14 MeV associated particle technique). The clear separation of the normalization and the shape in the present evaluation procedure is the result of the de facto separation of a constant factor and its uncertainty and the shape and its uncertainty in the experiments and in the data base.

The importance of the resolution in the interpretation of the reported experimental value is manyfold. The cross section is an average over the flux, $\phi(E)$,

$$\bar{\sigma} = \frac{\int_0^\infty \sigma(E)\ \phi(E)dE}{\int_0^\infty \phi(E)dE} .$$

The average energy should be defined as an average over the reaction rate,

$$\bar{E} = \frac{\int_0^\infty E\ \sigma(E)\ \phi(E)dE}{\int_0^\infty \sigma(E)\ \phi(E)dE} ,$$

or the neutron flux,

$$\bar{E} = \frac{\int_0^\infty E\ \phi(E)dE}{\int_0^\infty \phi(E)dE} .$$

and derived accordingly, but this is rarely the case. Even if $\bar{E}$ is derived from the reaction rate, $\sigma(E)\ \phi(E)$, a correction, k, is required to obtain from

$$\sigma(\bar{E}) = k\bar{\sigma}$$

the average cross section at the quoted average energy. The required correction is usually larger if $\bar{E}$ is derived as an average over the flux only ($\phi(E)$). Most frequently the correction was not applied and an uncertainty not accounted for.

The present evaluation is concerned with the evaluation of average (smooth) cross sections. However, fluctuations of the cross sections are known to exist and may influence the interpretation of individual experimental values. The fluctuations would have no impact on the evaluation of the average cross section if the data density were sufficiently high or the resolution poor. A correction factor can be calculated in order to obtain the smooth cross section from the cross section measured with the given resolution if the fluctuations are well enough known (from high resolution measurements). Another approach would be to account for the additional uncertainty.

2. Uncertainties of Experimental Values

Since the object of an evaluation is to provide the best value of a cross section which is derived from experimentally measured values, an uncerstanding and interpretation of experimental cross section values is required. It is assumed that the experimentally measured values have been reported in terms of the following parameters:

$\bar{E}$	The average energy at which it was measured,
ΔE	The uncertainty of this energy,
R	The energy resolution or energy spread,
σ	The cross section value,
$\Delta\sigma$	The total uncertainty of the measured value, and
$\Delta\sigma_{st}$	The statistical uncertainty.

More detailed information may be desirable, however, this appears to exhaust the information extractable from past experiments.

It is in the nature of the measuring process to be uncertain. The true experimental uncertainty is composed of several factors [1,7,13] and may be subdivided as follows:

$\Delta\sigma_R$	The reference uncertainty (mass, reference cross sections, etc.),
$\Delta\sigma_s$	The systematic uncertainty (estimated or calculated from the uncertainties of models and parameters used to calculate corrections, background subtraction, etc.),
$\Delta\sigma_{st}$	The statistical uncertainty (caused by the limited number of events counted),
$\Delta\sigma_a$	The accidental error, which may be revealed if the identical experimental is repeated (i.e. reproducibility),

$\Delta\sigma_u$ The unknown error, which is systematic in nature and caused by not recognizing necessary corrections or underestimating uncertainties.

$\Delta\sigma_{ps}$ The psychological error which is caused by satisfaction with agreement obtained with values reported by others, thus neglecting the search for additional effects in the measuring process or equipment which would require corrections, or, the opposite, that is the dissatisfaction with a disagreement with prior reported values and the subsequent search for one-directional corrections,

$\Delta\sigma_{pe}$ The personal error which is associated with the specific experimentor (Bessels observation of differences of experimental results obtained by leading astronomers of his time is noted in Ref. 13).

The reference uncertainty is usually beyond the responsibility of the experimentor. Sometimes it is possible to trace errors and improvements can be made on the reported measured value. The systematic uncertainty is the major responsibility of the experimenter. The statistical uncertainty is the best defined uncertainty and usually the only one the mathematical faculty bothers to deal with. Accidental errors could be limited to at best the statistical uncertainty by frequently repeating the experiment. However, experiments which involve long measuring times are not easily repeated. Some values can be recognized to have an accidental error, e.g., if the shape was measured and a singular point falls outside a range given by the statistics and known fluctuations of the cross section.

Quoting the existence of personal errors and psychological errors separated from the unknown error is of interest to gain some understanding of possible causes for unknown errors. In effect they affect the evaluation in a similar way as the unknown error though the psychological error might produce, as a side effect, historical trends in reported cross sections as shown in Ref. 1.

The quoted uncertainty, $\Delta\sigma$, is usually the square root of the sum of the squares of the reference uncertainty, the estimated or calculated systematic uncertainty of the experiment, the statistical uncertainty and sometimes the reproducibility (accidental error), i.e.

$$\Delta\sigma^2 = \Delta\sigma_r{}^2 + \Delta\sigma_s{}^2 + \Delta\sigma_{st}{}^2 \ (+ \Delta\sigma_a{}^2) \quad .$$

Whereas most experimenters now quote the uncertainty of their measurement this way, some quote only a partial uncertainty, for example, the statistical uncertainty and the systematic uncertainty, and thus give the impression of a higher accuracy of the measured quantity than actually achieved. The evaluator must take care to add the normalization uncertainty in order to weight different data on an equal basis.

The energy uncertainty, ΔE, and the resolution, R, result in cross section equivalent errors and uncertainties. The energy uncertainty may be subdivided in a similar way as the cross section uncertainty. One can translate ΔE into an additional cross section uncertainty, $\Delta\sigma_E$, by approximating

$$\Delta\sigma_E^{\,2} = \left(\frac{d\sigma}{dE}\right)^2 \Delta E2 + \left(\frac{d\sigma}{d\sigma_R}\right)^2 \left(\frac{d\sigma_R}{dE}\right)^2 \Delta E^2 + \sum_i \left(\frac{d\sigma}{dF_i}\right)^2 \left(\frac{dF_i}{dE}\right)^2 \Delta E^2 \quad ,$$

(linear regression), where the first term is the cross section uncertainty caused by the energy dependence of the measured quantity, the second term is that caused by the reference cross section and therfore does not apply to experiments which use absolute techniques or counters because they do not depend (in first order) on other cross sections. The last term accounts for the energy dependence of efficiencies and corrections which are part of deriving the quoted cross section. Often, the last term is small compared with the first two as a reuslt of the second order nature of corrections and the choice of flat-efficiency detectors. The second term is an important contribution which may reach or, in the case of the H(n,n) cross section, exceed the uncertainty of the reference cross sections itself which is accountable as $\Delta\sigma_R$.

3. The Handling of Discrepant Data Sets

Data are usually considered discrepant if they differ by more than their quoted uncertainties. An unknown error causes, with high probability, such discrepancy if it excéeds two or three standard deviations. The unknown error may be found in further analysis of the experiment by the experimenter or by others. If such is the case, a correction can be applied and the corrected value used in the evaluation. The values, thus entered, will still have unknown errors. If the number of independent data sets is sufficiently large, it is believed that an appropriate averaging will provide a better knowledge of the cross section and unknown errors might be assumed in individual experiments which disagree with the average outside the total estimated uncertainty of the experiment.[7] However, it must be relaized that the quoted total estimated uncertainty represents in most cases only a probability distribution and a criteria must be established for suggesting an unknown error and adding to the uncertainty. In the present evaluation a 2σ (2 standard deviations) criteria is used to double the given estimated uncertainty. Based upon the assumed normal distribution, this means that the probability for the quoted uncertainty of the experimental value to describe the real error of the vlaue is less than 2.3%.

Often data discrepancies are not a problem of one or two sets (which are assumed to be solvable with the 2σ criteria) but exist between several data sets on the one hand and several others on the other hand. The data are reanalyzed in such cases, searching for specific resolutions of the discrepancies. Corrections are applied and the data are re-entered into the evaluation if such solutions can be found. The resolution of the discrepancies by additional experimental work is recommended if solutions cannot be found by reanalysis. The discrepant data remain part of the data base until the ultimate resolution of such discrepancies. The evaluated result will be an average of the discrepant data and the error files will indicate the dispersion of the data. Such a procedure specifically avoids the assumption that only one of the two data sets is correct which would amount to taking a less than 50-50 chance of being totally right or totally wrong. The average of the discrepant data is at the largest probability for the true values, however, the implication of the existence of unknown errors is that the suggested solution may be half wrong or only half right. The acceptance of the average of discrepant data as part of the present evaluation procedure should reduce the cascading damage caused by a 50% probable totally wrong solution by one half.

III. THE EVALUATION OF ^{235}U(n,f) CROSS SECTION DATA

The present evaluation is an evaluation of the energy-averaged ^{235}U(n,f) cross section. It does describe gross structure (e.g. the step at 1 MeV, the bump at 2 MeV, the rise due to 2nd and 3rd chance fission). It does not describe "fine structure fluctuations" which are known to exist even above 0.1 MeV and it does not describe detailed features of some of the gross structure which might exist. However, such details of the structure are considered and included in the correction and interpretation of individual data points which are used in the evaluation of the average cross section.

1. The Present ^{235}U(n,f) Data File

The present data file consists of all data reported up to and including the Conference on "Neutron Physics and Nuclear Data for Reactor and Other Applied Purposes" which took place at Harwell in September 1978. Data reported earlier than the publication by White[14] were only included if reevaluations were available. It is assumed that older data would need substantial reevaluation efforts because data for neutron source reactions, cross sections and other critical values which might affect the results have undergone substantial changes. Moreover, the weight of these data would be very small as they were usually reported with uncertainties in the 5-10% range. Thus, the effort and costs of a major reevaluation of these older data seemed not to be justified. It should be noted, however, that older data are usually higher than the presently evaluated data and their inclusion would raise the present result. Some of the more prominent older data sets, though not included in the evaluation, are compared with the present result in Section IV.

The major part of the present data file is that documented in Ref. 15 but the file was updated to include newer data. The format of this file is standardized and is such that it permits its ready use in evaluations. Data measured relative to the hydrogen scattering cross section were updated to ENDF/B-V[16] values (Hopkins-Breit analysis[17]). Instead of using actual ENDF/B-V tabulated values it was found more convenient to use the expression

$$\sigma_H = 2.786 \cdot [1 + 0.2412 \cdot E + 0.0029 \cdot E^2]^{-1} + 17.68 \cdot [1 + 7.5644 \cdot E + 0.1534 \cdot E^2]^{-1}$$

which, up to 20 MeV, seems to describe ENDF/B-V within 0.15% or better.

A quantity of importance for the normalization of shape data which extend to the eV-energy range is the 7.8-11.0 eV fission cross section integral. Table I lists values which were referenced to the present thermal cross section of ENDF/B-V (583.5b). These values result in a weighted average of 243.1 ± 1.3 b-eV. One of the values measured by Gwin appears to be outside two standard deviations set as a limit in Section II.3. Doubling the uncertainty of this value results in an weighted average of 243.7 ± 1.0 b-eV. A more recent value measured by Wagemans[25] is 244.3 b-eV and a preliminary value by Brown[26] is 247.2 ± 4.9 b-eV. A fit of thermal parameters[43] resulted in a 0.8% higher thermal neutron fission cross section for ^{235}U which would increase the integral considered here. Recent findings[27] indicate that measurements using BF_3 counters may have resulted in low values, thus the weighted average would increase further. Until clarification of this latter effect and finalization of the data by Wagemans and by Brown, the value 243.7 ± 1.0 b-eV will be used for normalization.

Fluctuations in the $^{235}U(n,f)$ cross sections are well established below 100 keV and several data sets also indicate fluctuations in the cross section above 100 keV. Figure 1 compares data by Gayther[27] and Wasson.[28] A smooth cross section was subtracted in order to permit an easier comparison of the data. Though there seems to be agreement at some energies, there is disagreement at many others. Most dips in the data by Wasson[28] appear to be correlated with resonances in iron. A data set by Gwin[20] has some gross structure not seen in the other data sets. Because of these inconsistencies, corrections were not applied to any data. The analysis of cross section uncertainties caused by fluctuations as a function of the resolution of the experiment by Bowman[30] was used to estimate the additional uncertainties. Fluctuations of the cross section at higher energies were reported by Albert[31] but could not be verified.[32,33] More recently, measurements by Osterhage et al.[34] show large fluctuations, but it appears that these data suffer from substantial statistical uncertainties and energy scale problems.

The data sets used in the present evaluation are given in Appendix I. Renormalizations due to changes of the reference cross section, fission integral and fission masses were applied to the data sets resulting in permanent changes. Additional uncertainties due to energy uncertainties and cross section fluctuations were taken care of during the different stages of the evaluation.

TABLE I. The Fission Integral from 7.8 to 11.0 eV

Author	Reference	Value/B-eV
Deruytter, Wagemans	18[a]	243.07 ± 2.4
Czirr	19	243.9 ± 1.9
Gwin	20[a]	235.92 ± 3.5
Gwin	21	245 ± 3
ORNL/RPI	22[a]	241.3 ± 4.8
Bowman	23[a]	251.91 ± 7.6

[a]Values updated as quoted in Ref. 24.

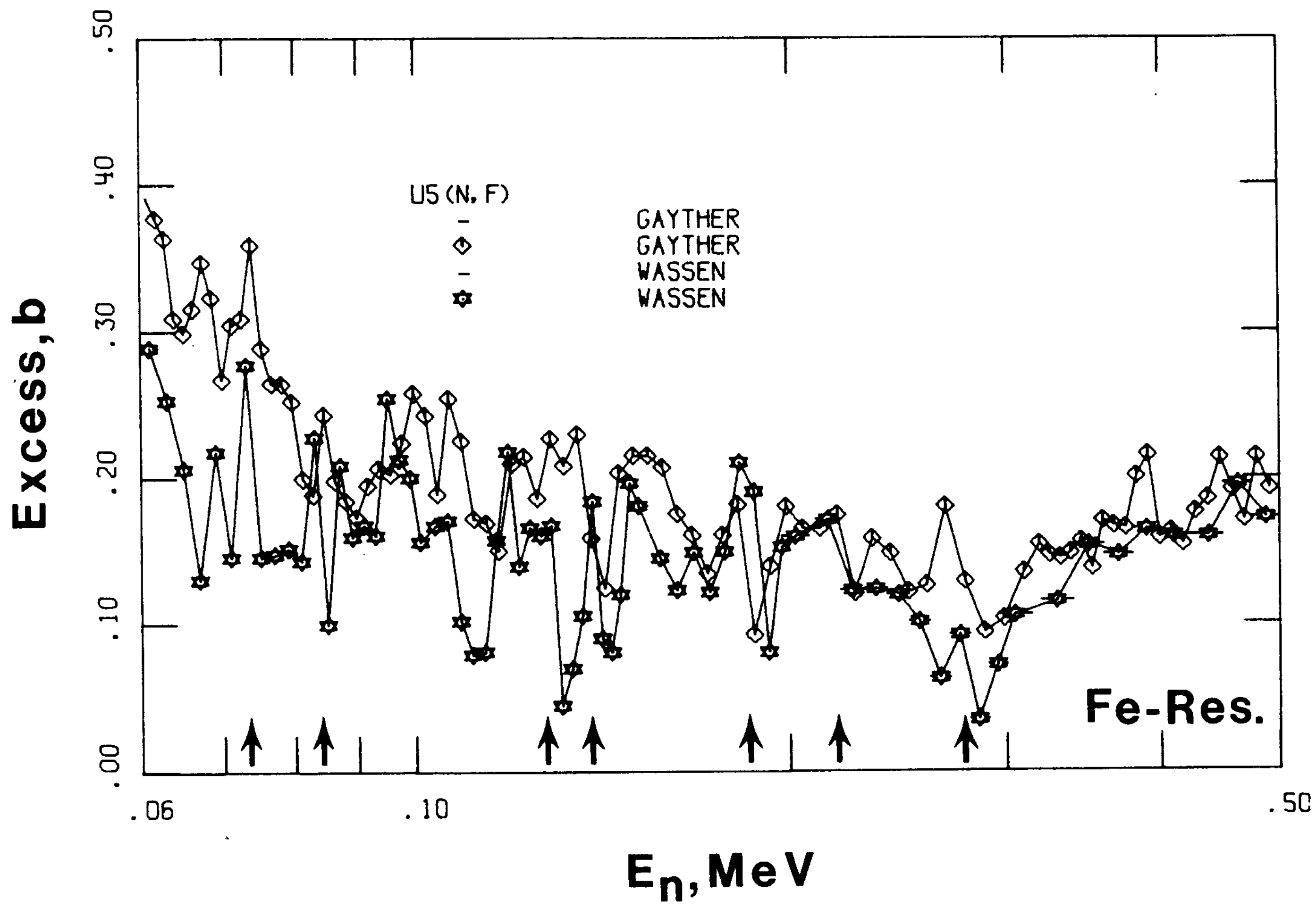

Fig. 1. Comparison of the Structure Obtained by Wasson[28] and Gayther[27] in the Energy Range from 60 keV to 500 keV. A smooth "background" cross section was subtracted in order to better facilitate comparison.

TABLE II. Shape Evaluation Input

Set	Set Name	Energy Range MeV
1	Smith[29]	2.2 - 20.5
2	Barton[40]	1.0 - 6.0
33	Czirr[38], Combined Set	0.75 - 20.0
4	Kari[39]	1.0 - 20.0
51 and 52	Szabo[44]	0.01 - 5.5
6	White[14]	0.04 - 14.1
71	Poenitz I[41,42]	0.2 - 8.2
72	Poenitz II[42]	0.03 - 3.5
8	Gayther[29]	0.001 - 0.95
10	Wasson[28]	0.005 - 0.75
111 and 112	Kaeppeler[37]	0.4 - 1.0
17	Wasson and Meier[45]	0.25 - 1.2
153	Carlson, Combined Set	1.2 - 0.2
91	U. Michigan	0.14 - 0.96

2. The Evaluation of the Cross Section Shape

Table II indicates the data sets which were used in the calculation of the presently evaluated shape. Data were not modified beyond the permanent modifications indicated in the file and the modifications to the uncertainties discussed below. An energy grid with 63 values was established for use in the shape evaluation. This grid was found sufficient to describe all gross structure of the average cross section between 0.1 and 20.0 MeV. Experimental data points were extrapolated to the grid energy, E_i, if they were found in the interval $[E_i - 0.5\ (E_i - E_{i-1});\ E_i + 0.5\cdot(E_{i+1} - E_i)]$ using the shape of the cross section in the range $[E_{i-1}, E_{i+1}]$. This shape was obtained from another evaluation.[5]

Interpolation was attempted by a power curve, a 2nd-order polynomial, and a linear curve. Finally, a power curve was used below 1 MeV and linear interpolation above. Little difference for any choice of curves of interpolation can be expected for a sufficiently small grid and reasonably random (in E) distributed data points. The choice of the cross section used in this extrapolation procedure does not influence significantly the outcome of the shape evaluation.

The uncertainty of the value at a grid point was determined from a variety of components. The total uncertainty of the experimental points was increased by the uncertainties caused by the uncertainty in energy (see Sect. II.2) and the uncertainty due to cross section fluctuations. The former requires that the energy uncertainty be known, the latter requires that the resolution be known. For experiments which did not quote the energy uncertainty it was assumed to be 20% of the resolution for Van de Graaff type experiments and 10% of the resolution for Linac type experiments. The resolution was assumed to be 1% of the energy for Linac type experiments and 5% for Van de Graaff type experiments if it was not given. The uncertainty of the grid point was reduced by part of the statistical uncertainty if several experimental data points contributed to it. For this purpose it was assumed that the statistical uncertainty originated from a limited number of events detected in the experiment. The uncertainty of the grid point is reduced by the normalization uncertainty which applies only to absolute measurements. An additional uncertainty of the grid point is that caused by the procedure in obtaining it, which was estimated to be 0.5%.

The values obtained from one data set provide cross section ratios between any two of the energy grid points for which they are available. In forming the ratios, R_{ik}, between the cross section values, σ_i, σ_k from these grid points the arbitrary or absolute normalization of the data set is removed. The different experiments provide contributions to some of the R_{ik}'s which yield a weighted average $\overline{R}_{ik}$ with an uncertainty ΔR_{ik}. By summing (weighted) the contributions to any σ_i, one obtains a system of equations which can be resolved in a "roll-back" procedure by defining a starting point $\sigma_1 = x$, where x is arbitrary. The roll-back may as well be started on the high energy side, that is, with setting $R_{63} = y$, where y again is arbitrary. Results starting from different sides will be expected to differ somewhat, reflecting data uncertainties and inconsistencies. Figure 2 shows the results obtained from the shape evaluation with the techniques described above. At this stage an eyeguide curve was drawn through the calculated values which represents the expected physical behavior of the cross section better. The paper by Moore[36] gave guidance over most of the energy range. The origin of the step at 1 MeV is well understood[37], however, its detailed structure is unknown and difficult to extract from the available data. In this range the largest difference occurs between the two calculated shape data sets, it is due to the weight of Czirr's shape data[38] which suggest a much larger step than all other data sets. The fourth chance fission threshold did not show up in the calculated values because of the contradictory results of the data by Kari[39] and by Czirr[38] above 18 MeV and was introduced at this stage.

3. Evaluation of the Normalization

The weighted average of factors obtained from the different absolute data sets was used for the normalization of the arbitrarily normalized shape curve. These factors are given in Table III and were derived as the weighted average of the ratios of the data and the values of the shape curve. The total uncertainties were used to determine the weights. The cross section uncertainties caused by energy uncertainties and cross section fluctuations were added. The uncertainty of the shape curve was also taken into account. The uncertainty of the normalization factor was determined as the average of the uncertainties for the individual ratios and reduced by part of the statistical uncertainty if more than one value contributed. Table III shows in one column all normalization factors and in another column a selection of such factors. In the latter, the three Russian sets (16, 12, 14) were excluded because these data are not well documented and do not permit a detailed analysis. the measurement by Alrasov may be superceded by the work by Arlt. An "Information-Publication" by the Technical University of Dresden shows Alrasov as a coauthor for the measurement reported by Arlt et al. The shape of the data by Barton et al.[40] shows several features which cannot be seen in any other data set (for example, a very flat shape between 1 and 2 MeV, a "step-up" around 3 MeV). Therefore, only the 3 MeV value was used from this data set to determine the normalization factor. The value at 3 MeV is the result of many consistency-check measurements described by Barton et al.[40] The BND data published in 1976 by Poenitz[41] were used in the selected set because the BND data[42] from 1974 are not totally independent from the VSO_4-bath measurement. The data by Kaeppeler were excluded from the column of selected values because the shape appears not to agree with any other measured shape. Finally, the Cf-source average value from the University of Michigan was excluded because it is not independent from the photo-neutron source measurements by the University of Michigan and the neutron source strength value used by Heaton et al.

The weighted and the unweighted average of the normalization factors agree very well, showing that no bias was introduced by data with lower uncertainties. The average from the column of selected values given in Table III is used in the present evaluation for the normalization of the shape curve. Table IV shows the average differences of the absolute data

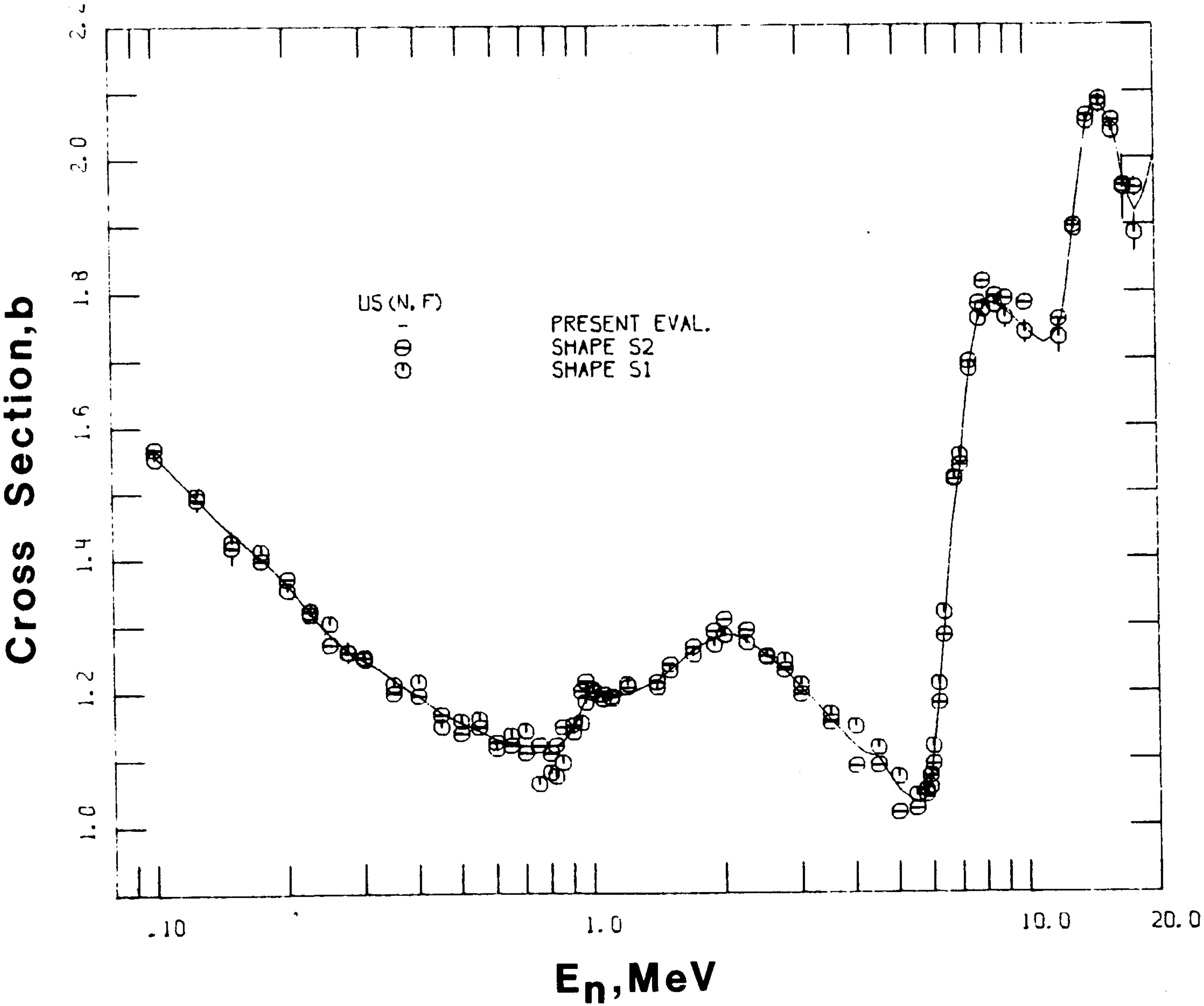

Fig 2. Comparison of the Shape of the Present Evaluation with the Actually Calculated Values. Shape S1 was obtained by starting at 20 MeV and S2 by starting at 0.1 MeV.

TABLE III. Normalization Factors

Set	Set Name	All Normal Factors	Selected Factors
1	Smith	1.006 ± 0.066	1.006 ± 0.066
2	Barton	1.012 ± 0.029	1.003 ± 0.029
71	Poenitz BND	1.008 ± 0.030	0.988 ± 0.027
74	Poenitz VSO_4	1.003 ± 0.038	1.003 ± 0.038
73	Poenitz AA	1.002 ± 0.038	1.002 ± 0.038
4	Kari	1.048 ± 0.039	1.048 ± 0.039
16	Alkharov	1.060 ± 0.035	-
12	Kuks	1.051 ± 0.041	-
18	Cance	1.007 ± 0.038	1.007 ± 0.038
6	White	1.032 ± 0.042	1.032 ± 0.042
17	Wasson, Meier	0.986 ± 0.039	0.986 ± 0.039
52	Szabo	1.012 ± 0.032	1.012 ± 0.032
111	Kaeppeler	1.054 ± 0.037	-
91	U. Midrigan	1.019 ± 0.024	1.019 ± 0.024
10	Wasson	0.991 ± 0.036	0.991 ± 0.036
251	Arct	1.004 ± 0.032	1.004 ± 0.032
14	Adamov	1.048 ± 0.021	-
92	U. Michigan Cf	1.006 ± 0.021	-
13	Heaton	1.001 ± 0.025	1.001 ± 0.025
Unweighted Average		1.018	1.007
Weighted Average		1.019 ± 0.005	1.006 ± 0.004

TABLE IV. Average Differences of Data Sets from the Present

Set	Set Name	0.1 - 20.0 MeV	0.1 - 1.0 MeV	1.0 - 1.0 MeV
1	Smith	1.002 ± 0.058	-	1.002 ± 0.058
2	Barton	1.010 ± 0.021	-	1.010 ± 0.021
71	Poenitz BND	0.989 ± 0.020	0.998 ± 0.019	0.984 ± 0.021
74	Poenitz VSO_4	0.996 ± 0.037	0.996 ± 0.037	-
73	Poenitz AA	0.997 ± 0.037	0.997 ± 0.037	-
4	Kari	1.040 ± 0.029	-	1.040 ± 0.029
16	Alkharov	1.053 ± 0.034[a]	-	1.053 ± 0.034[a]
12	Kuks	1.045 ± 0.038	-	1.045 ± 0.038
18	Cance	1.000 ± 0.023	-	1.000 ± 0.023
6	White	1.032 ± 0.034	1.017 ± 0.042	1.039 ± 0.027
17	Wasson, Meier	0.980 ± 0.035	0.980 ± 0.035	0.978 ± 0.036
52	Szabo	1.005 ± 0.027	1.018 ± 0.027	0.998 ± 0.028
111	Kaeppeler	1.046 ± 0.033	1.053 ± 0.032	1.037 ± 0.033
91	U. Midrigan	1.013 ± 0.021	1.013 ± 0.021	-
10	Wasson	0.985 ± 0.035	0.985 ± 0.035	-
251	Arlt	0.998 ± 0.011	-	0.998 ± 0.011
14	Adamov	1.042 ± 0.030[a]	-	-
92	U. Midrigan	1.000 ± 0.014	-	-
13	Heaton	0.995 ± 0.020	-	-
Weighted Average		1.006 ± 0.004	1.008 ± 0.007	1.008 ± 0.006

[a]Uncertainty doubled based upon the 2σ criteria.

sets from the present evaluations. The reason the weighted average is not 1.000 is due to the selection made in Table III but not made in Table IV and the accounting of the shape uncertainty in Table III and not in Table IV. The most important result shown in Table IV is that the choice of different energy ranges would not result in different normalizations. The change of the normalization due to inclusion of the ^{252}Cf fission spectrum averaged cross sections is only 0.2%.

IV. EVALUATION RESULT AND COMPARISONS

Table V gives the result of the present evaluation of the ^{235}U cross section which will be used in a later consistency fit with other data as input for ^{235}U. Figure 3 compares the present result with the Evaluated Nuclear Data File ENDF/B IV. The present result is lower over most of the energy range and substantially so between 1.0 and 1.5 MeV and above 13 MeV. The difference above 13 MeV is the result of data inconcsistencies in older data and the improvement provided with more recent measurements not available for ENDF/B-IV but used in the present evaluation. The difference between 1.0 and 1.5 MeV is of importance in the interpretation of the average of the ^{235}U cross section over a fission spectrum and for the interpretation of critical test facilities with a hard spectrum, as for example GODIVA. It appears that this difference is due to the selection of one experimental data set for ENDF/B-IV where many were available which predicted a different shape between 1 and 2 MeV. Figures 4-11 compare the present evaluation result with the absolute data used in the evaluation. Figures 12 and 13 compare the shape of the present evaluation with shape data which were normalized to the evaluated values. Figure 14 shows the data by Kari[39] and by White[14] normalized to the present result. This shows that these data agree rather well with the shape of the present evaluation but differ in normalization. Figures 15-17 show the energy range from 1 to 6 MeV in which a discrepancy exists between the data by Barton et al.,[40], Kari,[39] Czirr and Sidhu[38] on the one hand and Szabo,[44] Poenitz,[41] and Carlson and Patrick[46] on the other hand. The discrepancy is mainly in shape between 2 MeV and 6 MeV, except for the data by Kari which differ over the whole range in normalization and are between the two discrepant sets in shape. It is interesting to note that the measurements by Czirr and Sidhu[38] and by Carloson and Patrick[46] used the same basic experimental procedure but differ substantially in shape. Figure 18 compares the evaluation result with some older data sets not included in the present evaluation. Prominent data sets which had influenced evaluated nuclear data files up to ENDF/B-III (Allen and Furgeson,[47] Henkel,[48] and Diven[49]) are 8% higher than the present evaluation result but appear to agree reasonably well in shape as can be seen in Fig. 19, where data from Fig. 18 were multiplied by a constant factor to better agree in magnitude with the present evaluation.

The average of the present evaluated ^{235}U(n,f) cross section over the ^{252}Cf spontaneous fission neutron spectrum is 1.215 b. This is to be compared with measurements by Heaton et al.[50] and by the University of Michigan.[51] The experimental values of (1.208 ± 0.024) and (1.215 ± 0.022) are in good agreement with the present result, however, a measurement reported by Abramov et al.[52] yielded (1.266 ± 0.019)b, which suggests a substantially higher

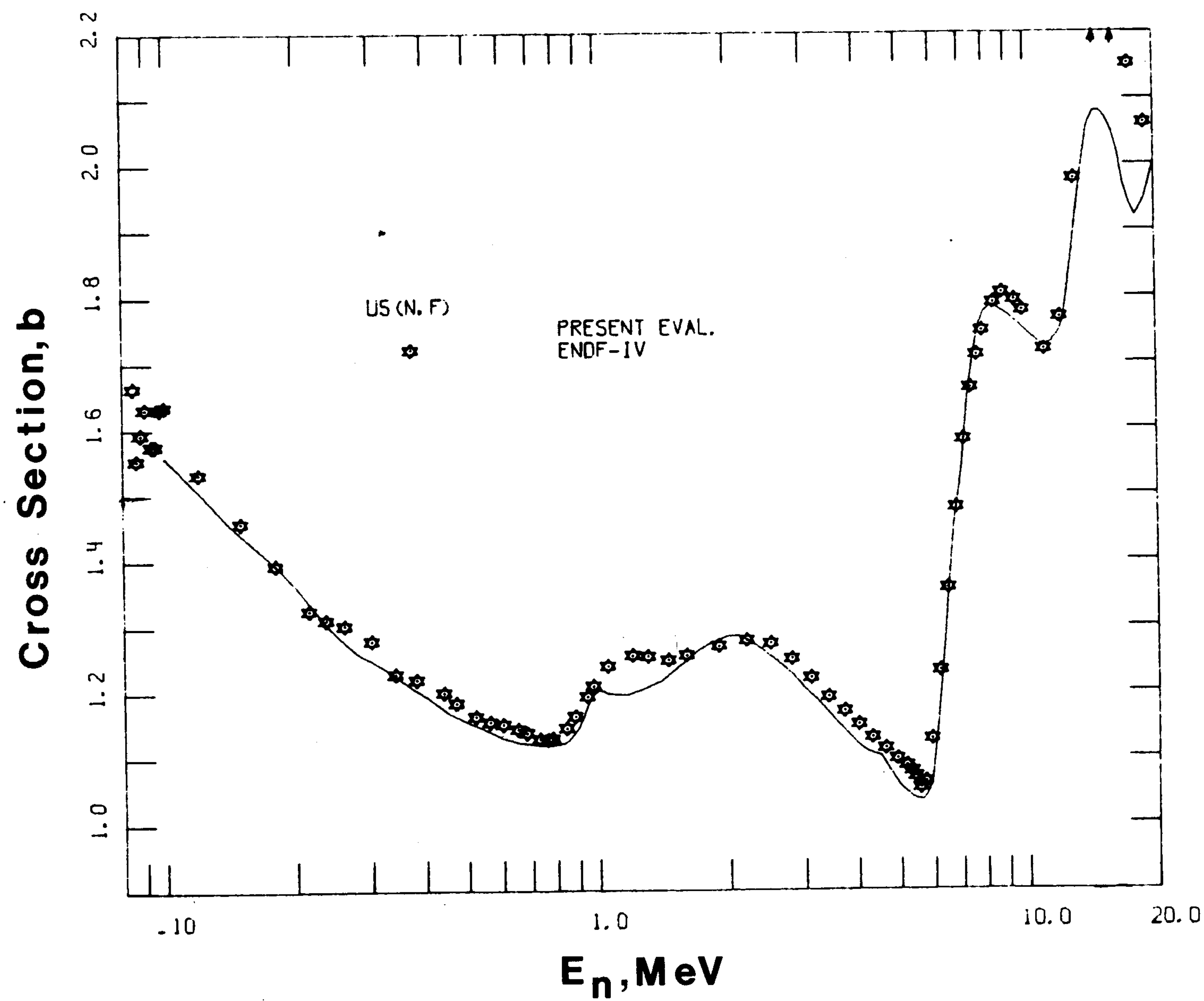

Fig. 3. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with ENDF/B-IV.

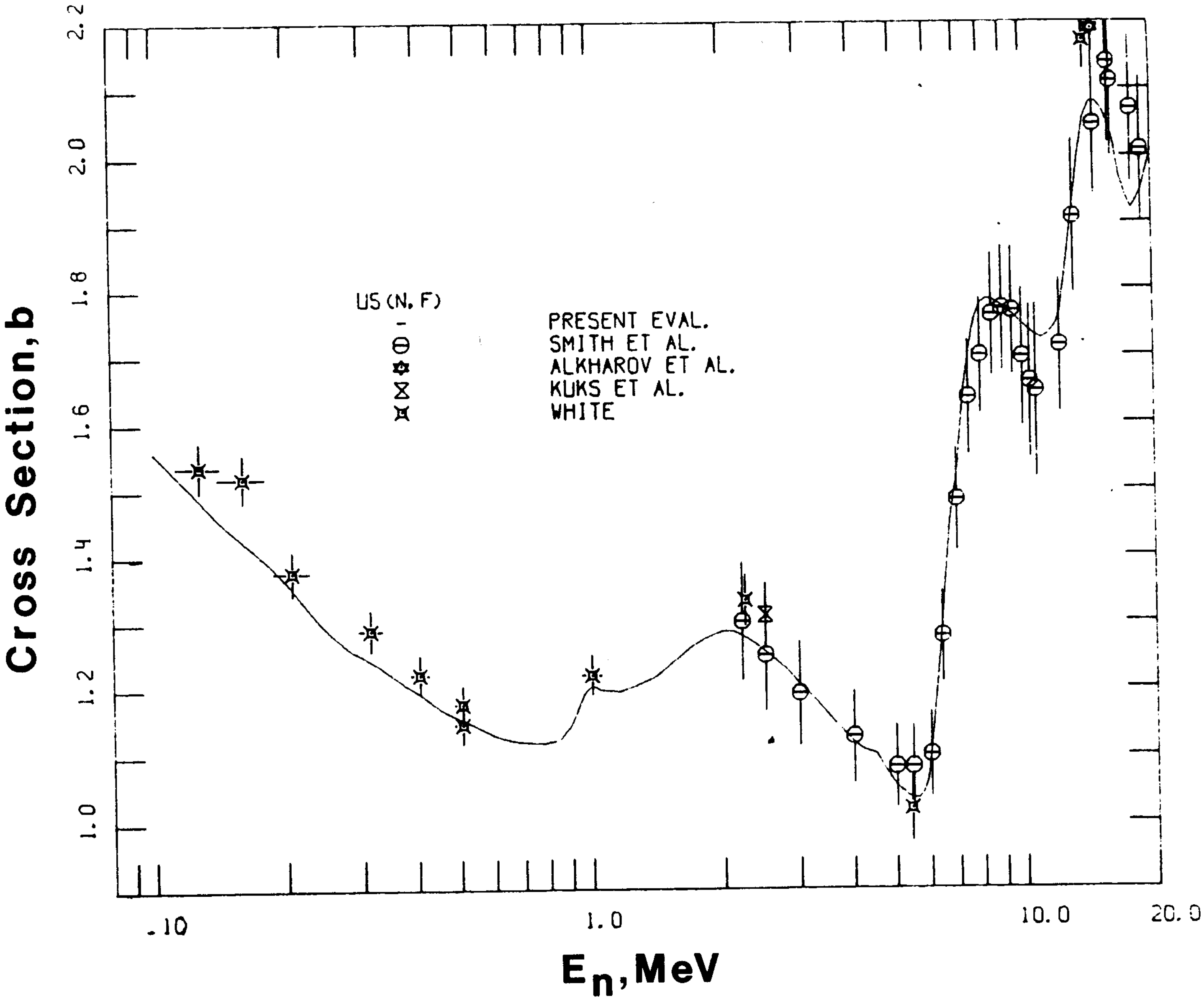

Fig. 4. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with Experimental Data.

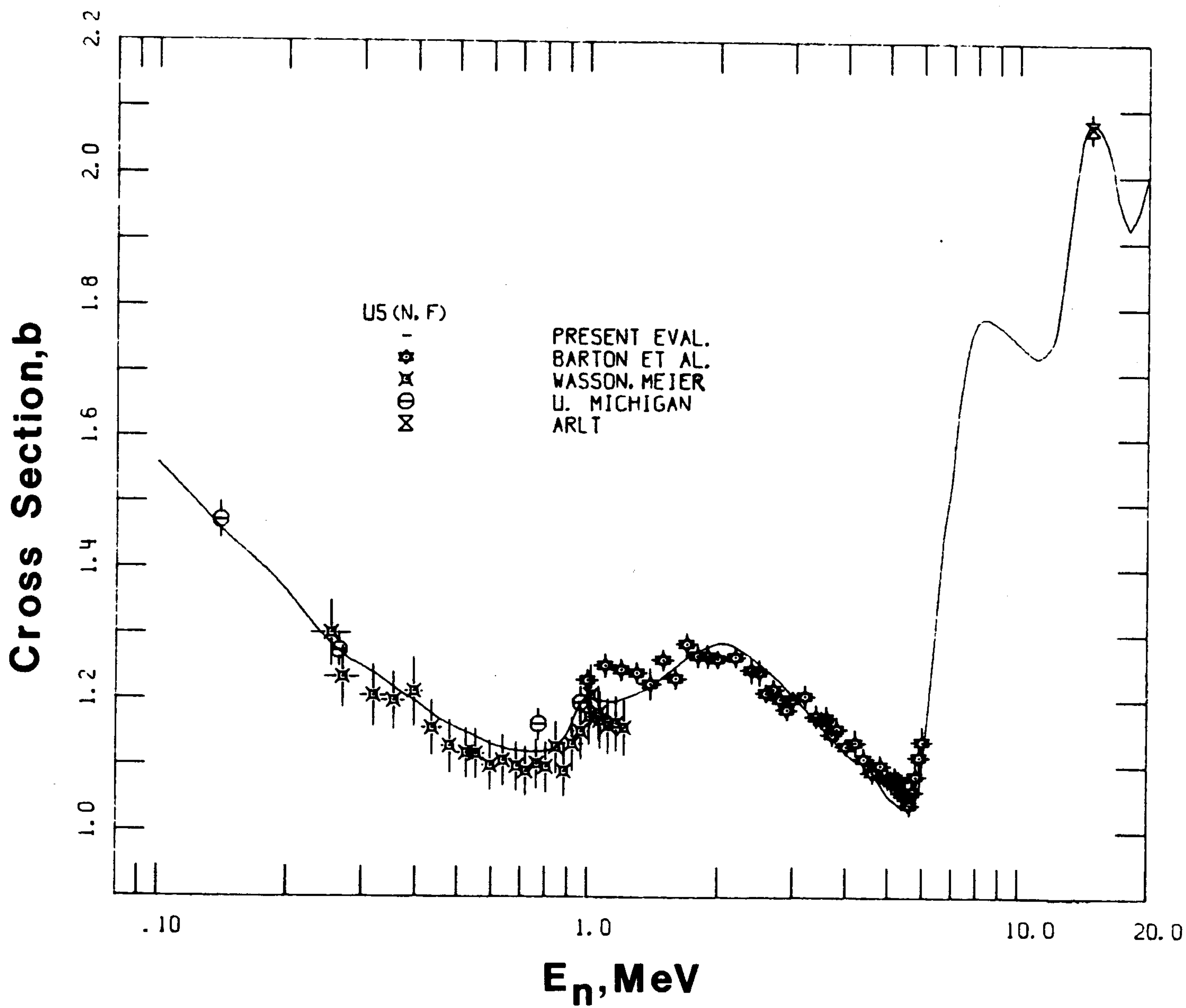

Fig. 5. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with Experimental Data.

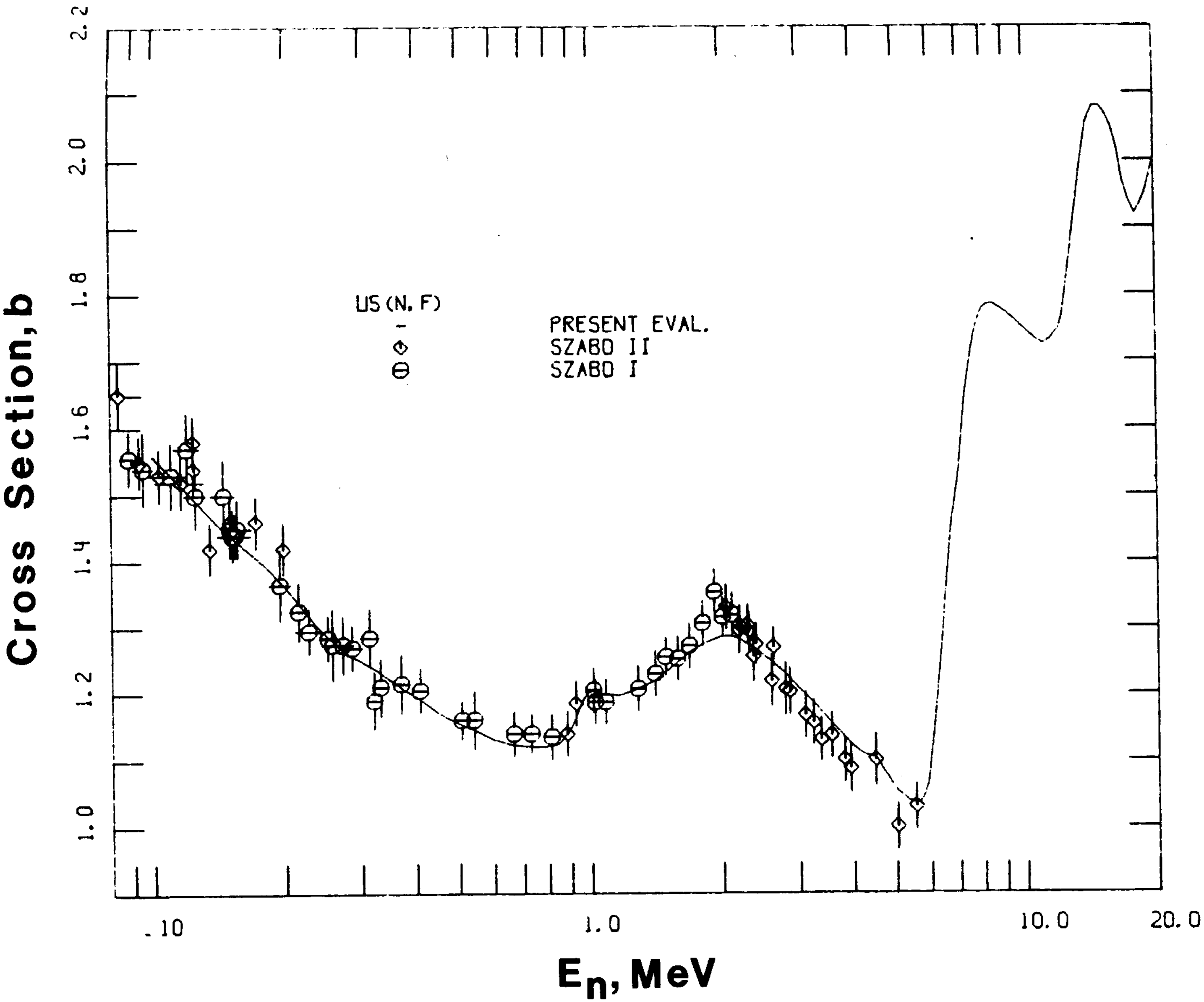

Fig. 6. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with Experimental Data.

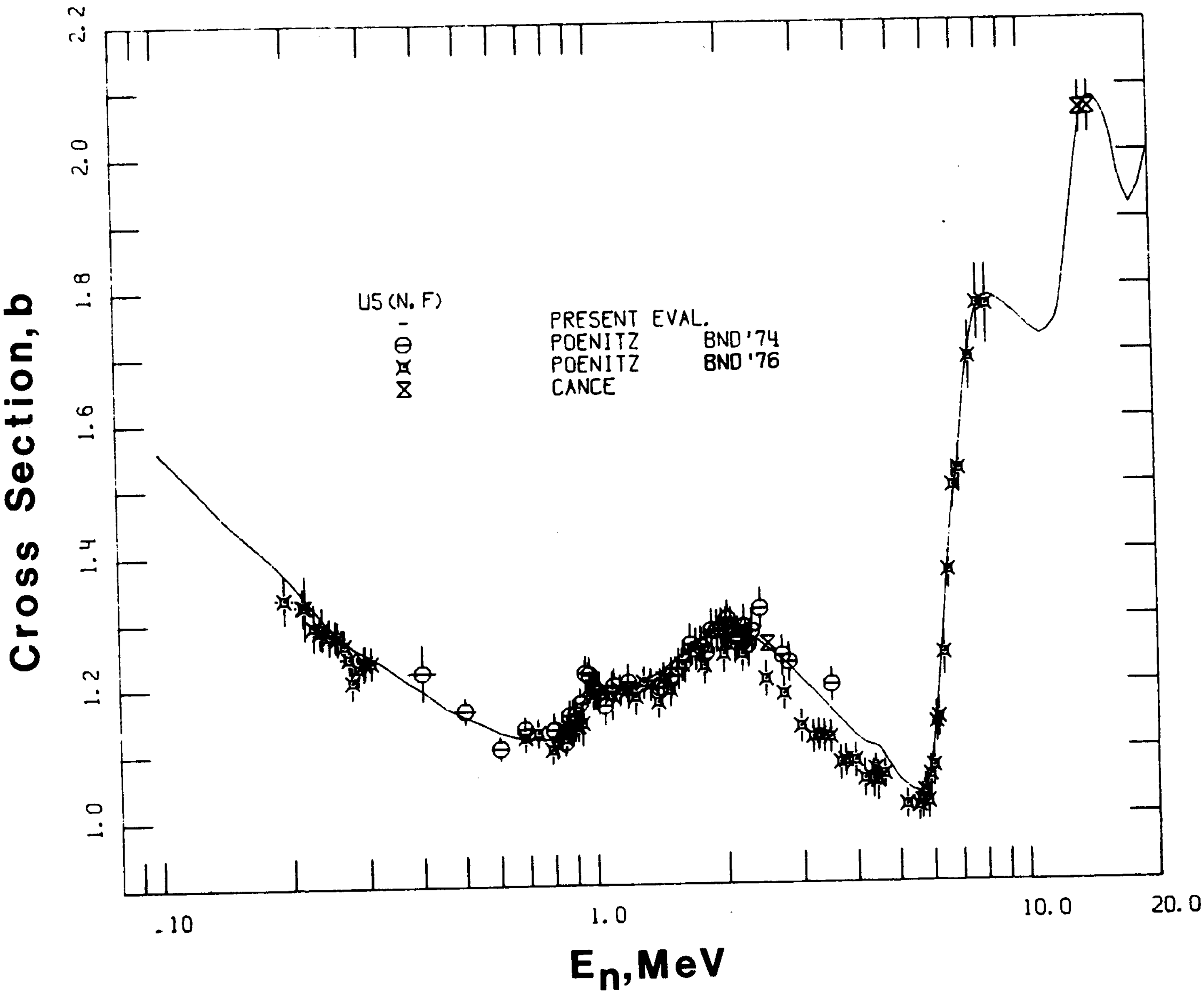

Fig. 7. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with Experimental Data.

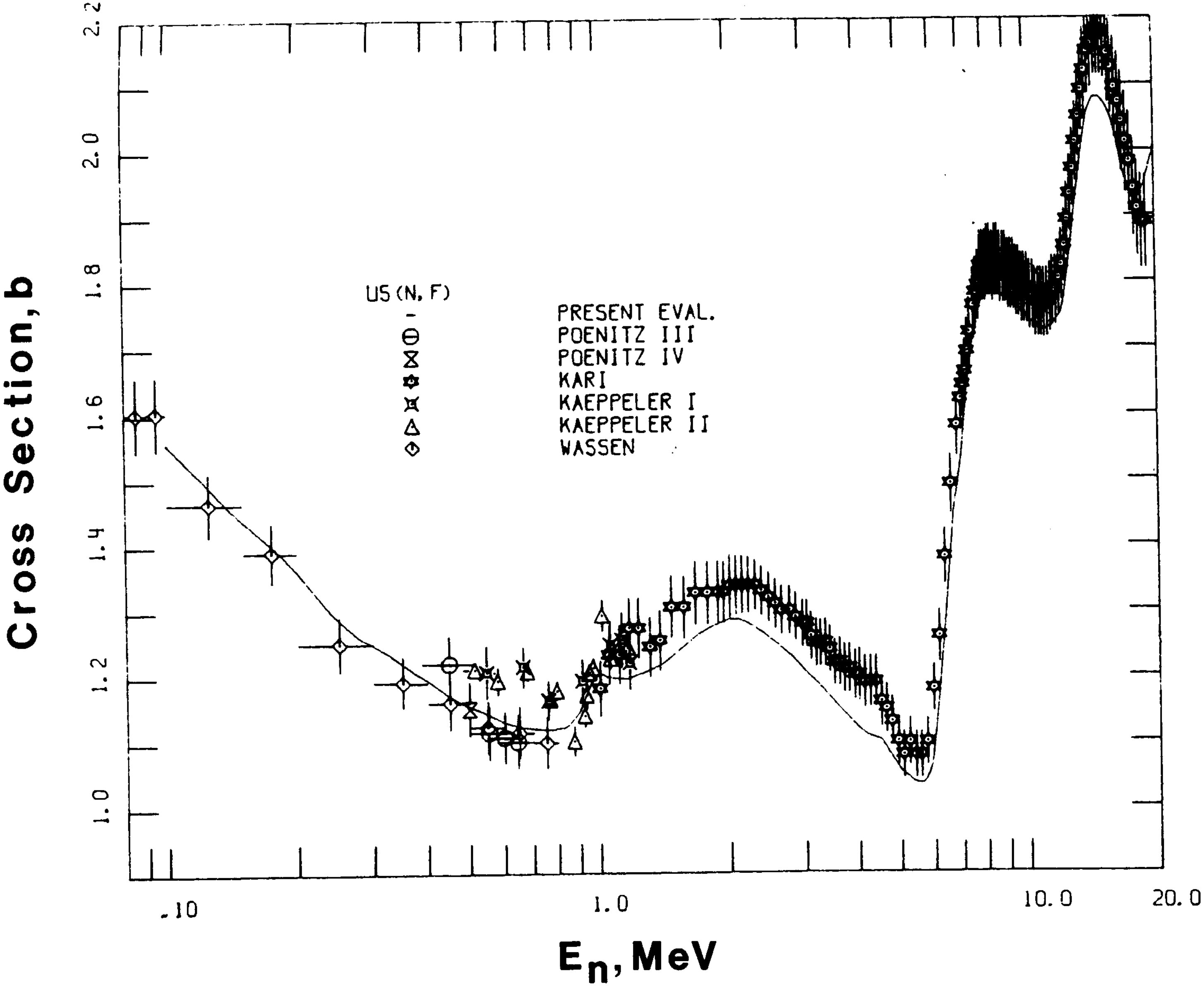

Fig. 8. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with Experimental Data.

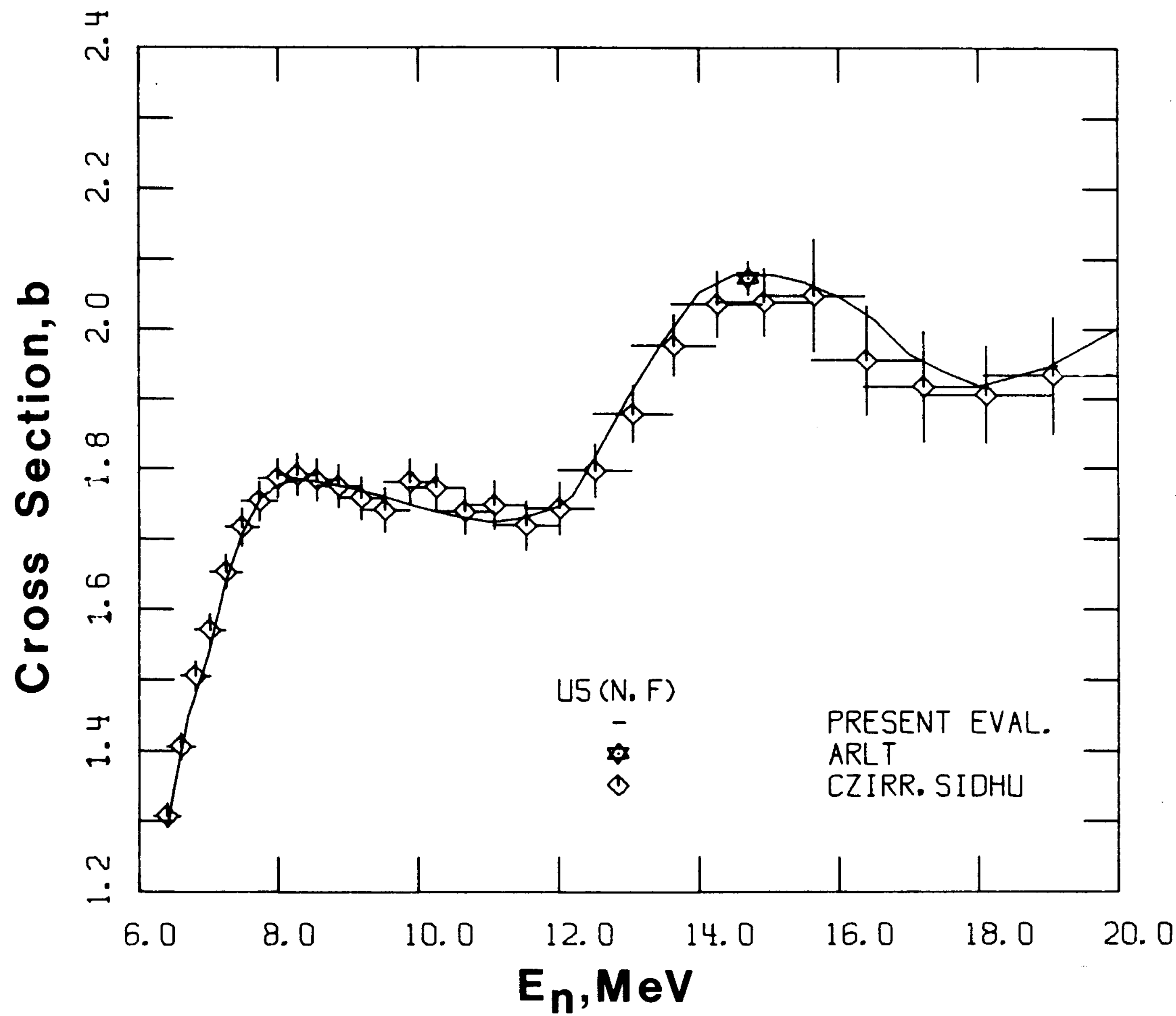

Fig. 9. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with Experimental Data.

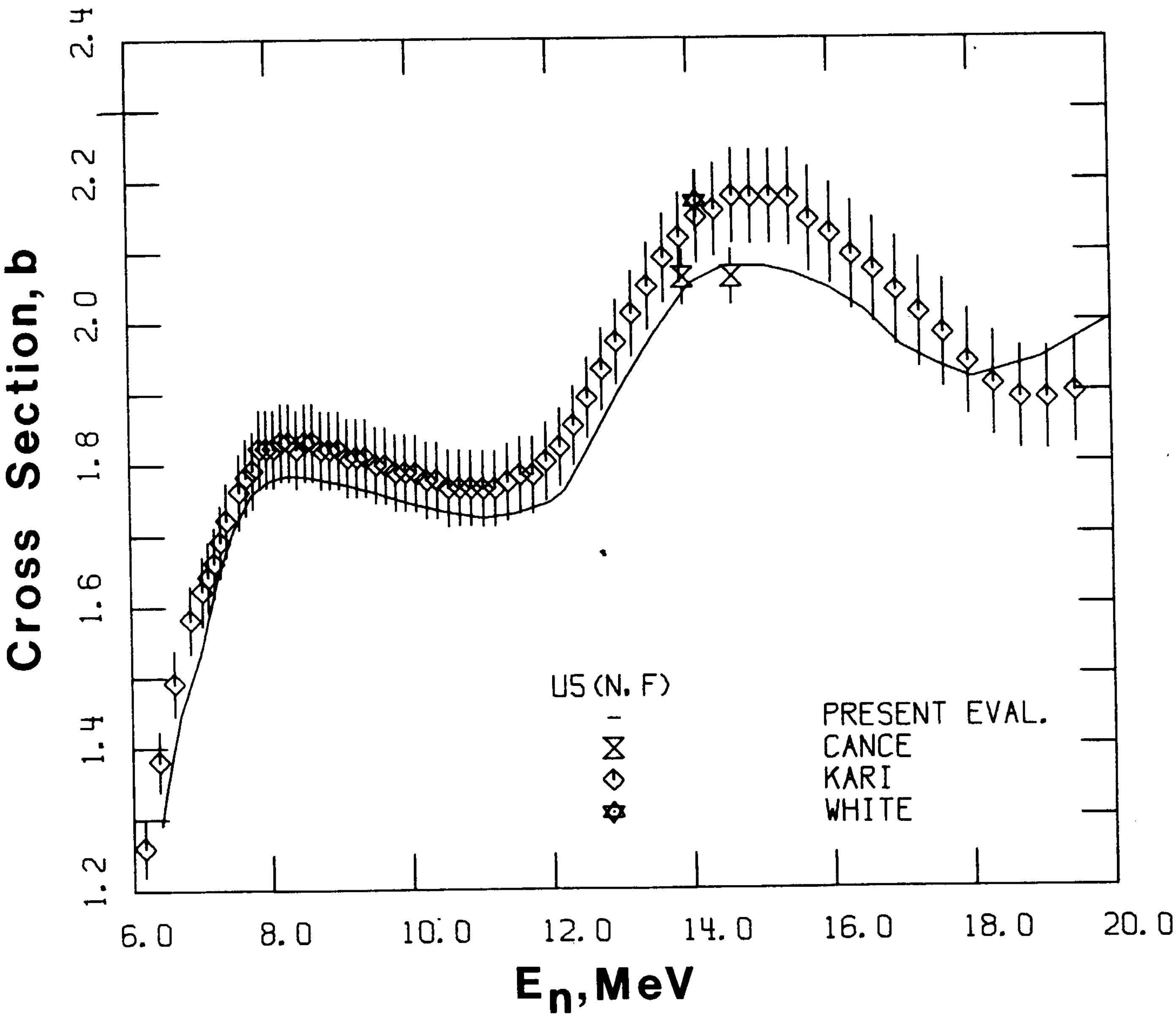

Fig. 10. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with Experimental Data.

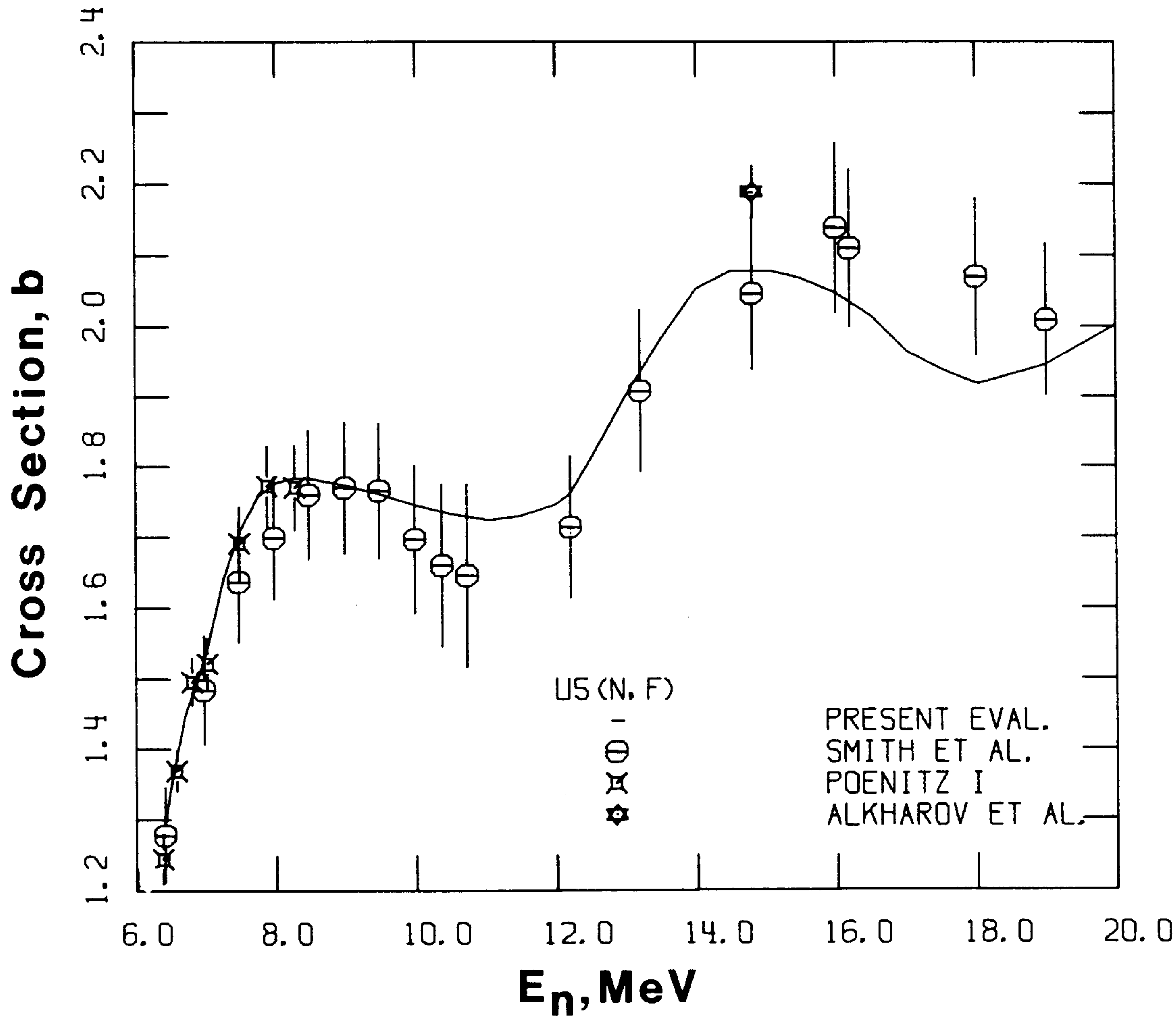

Fig. 11. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with Experimental Data.

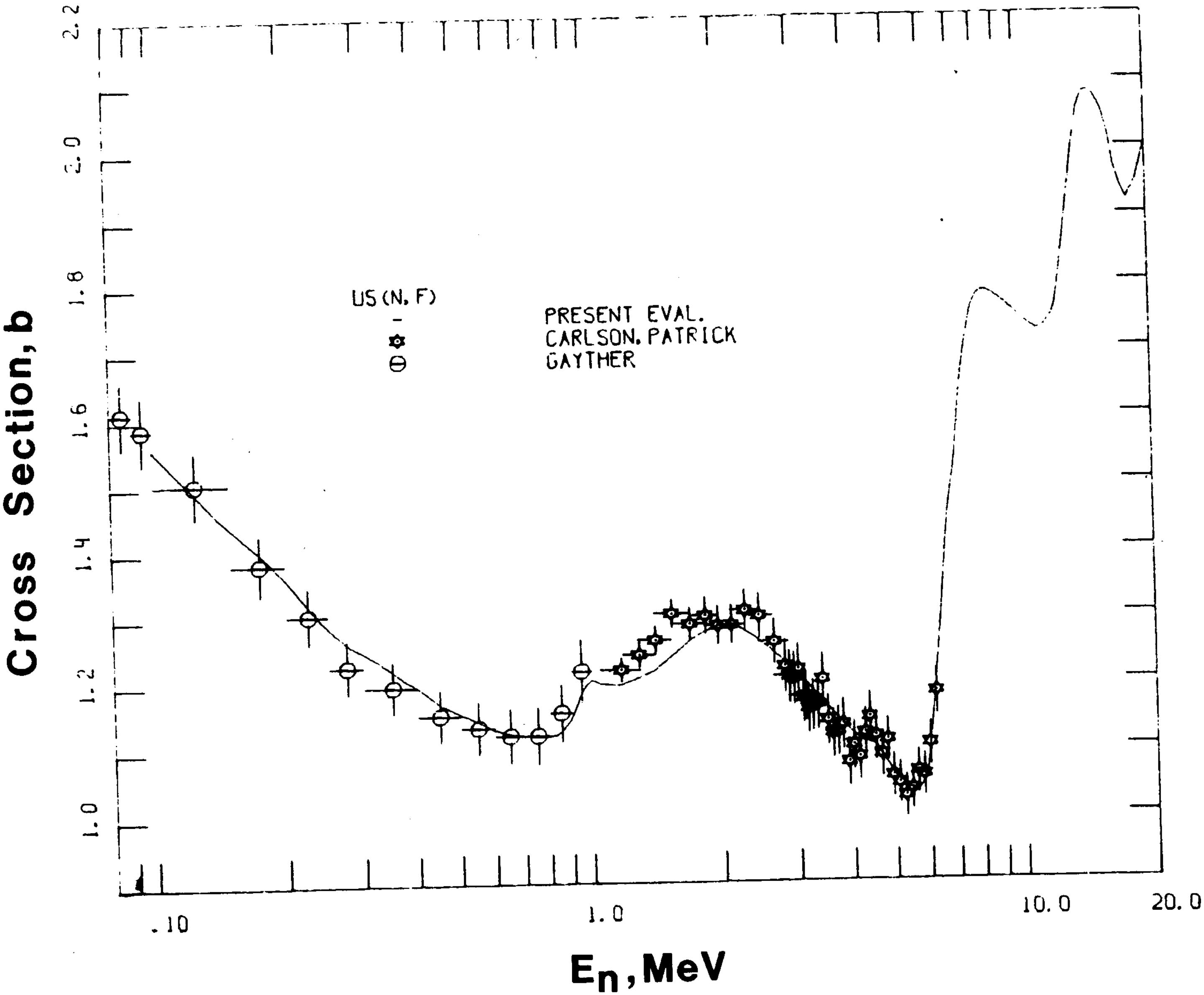

Fig. 12. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with Experimental Data.

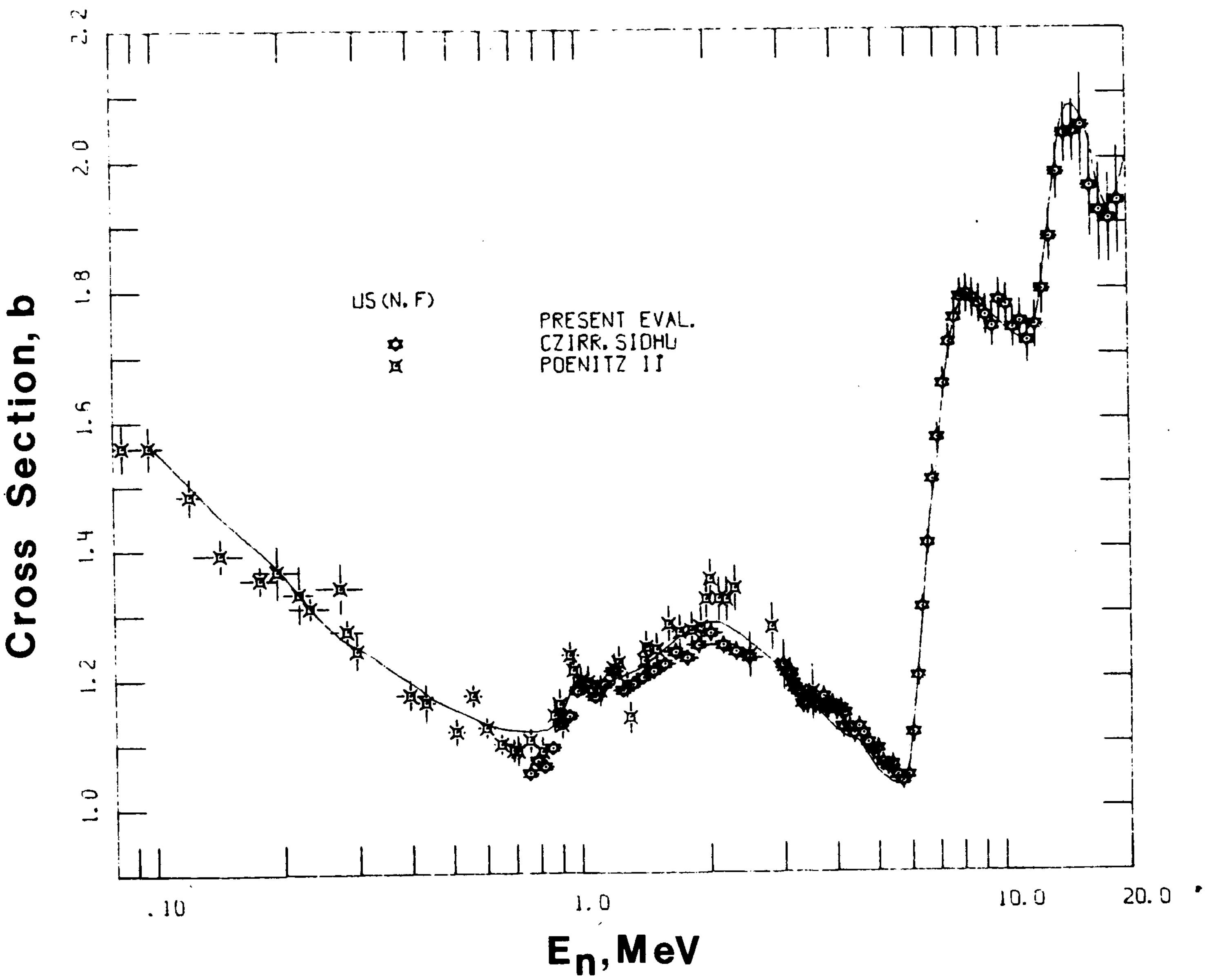

Fig. 13. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with Experimental Data.

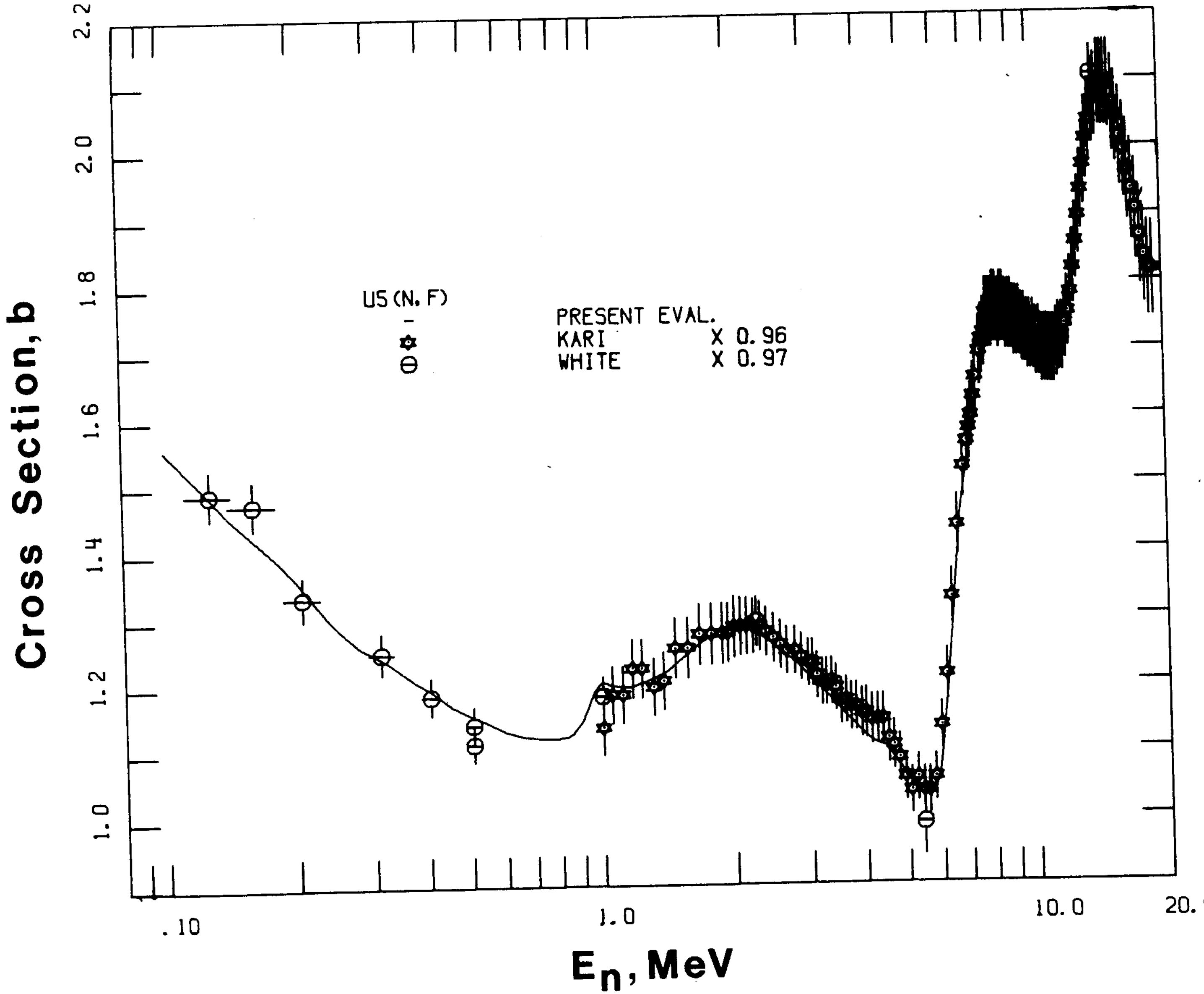

Fig. 14. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with Experimental Data.

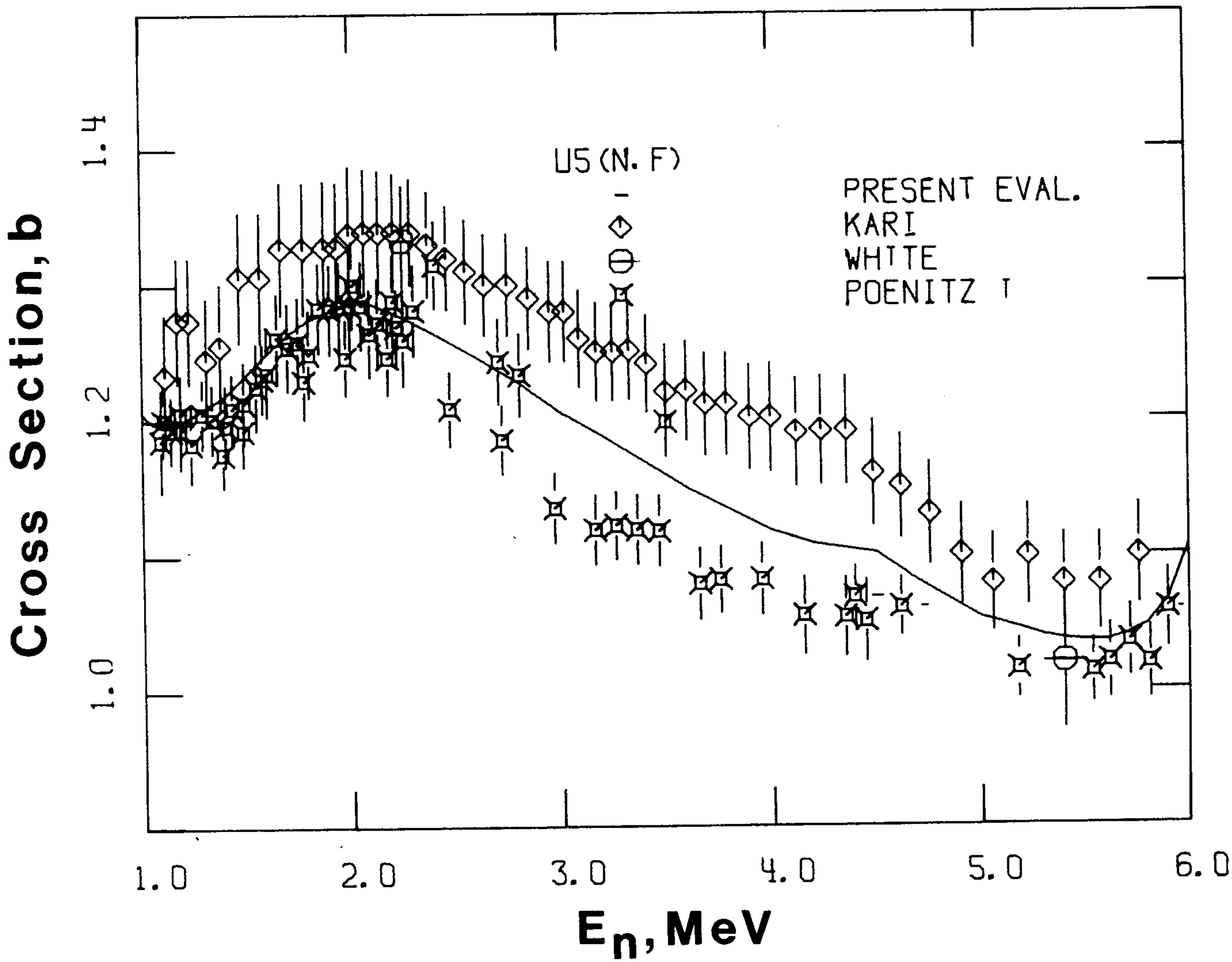

Fig. 15. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with Experimental Data.

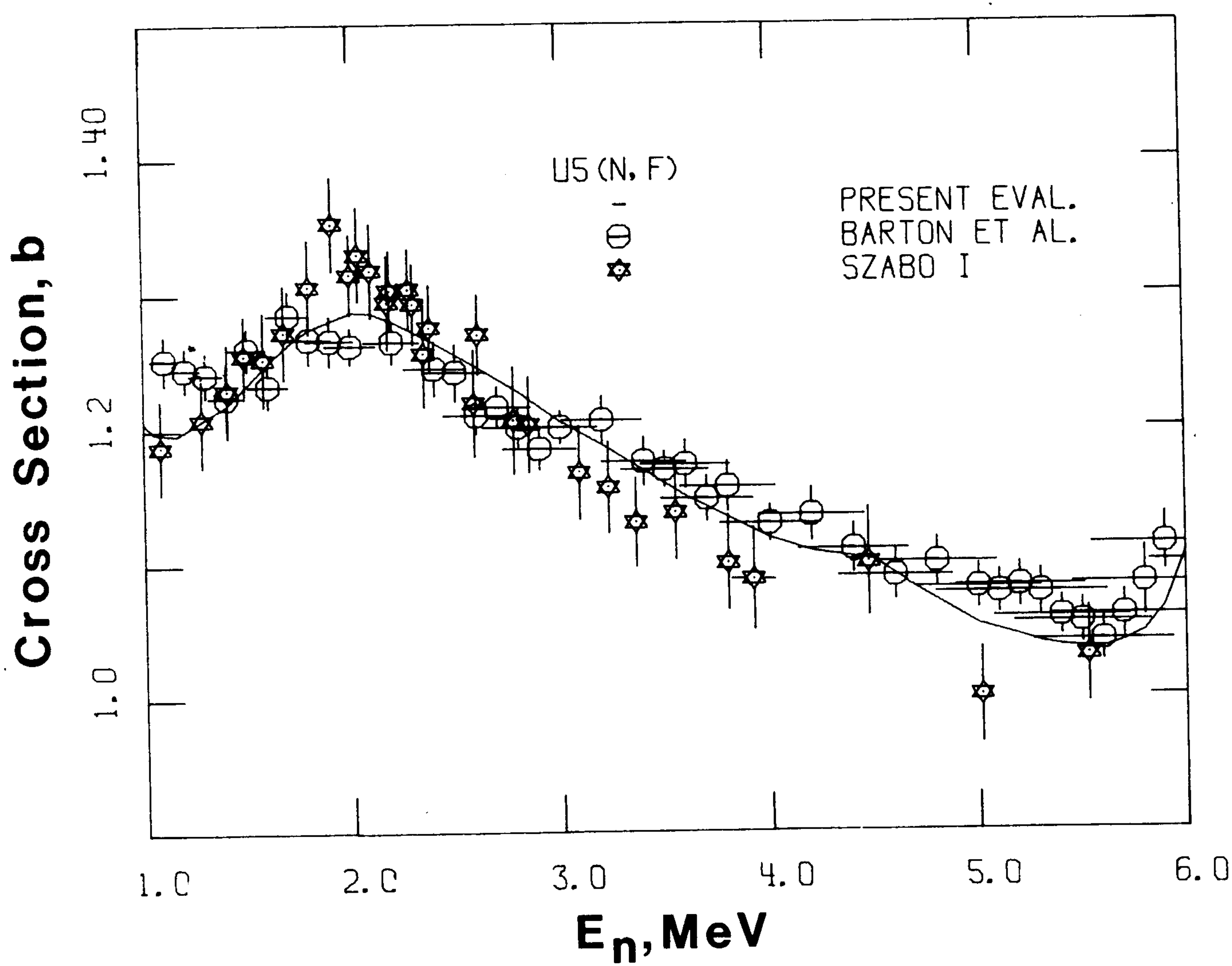

Fig. 16. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with Experimental Data.

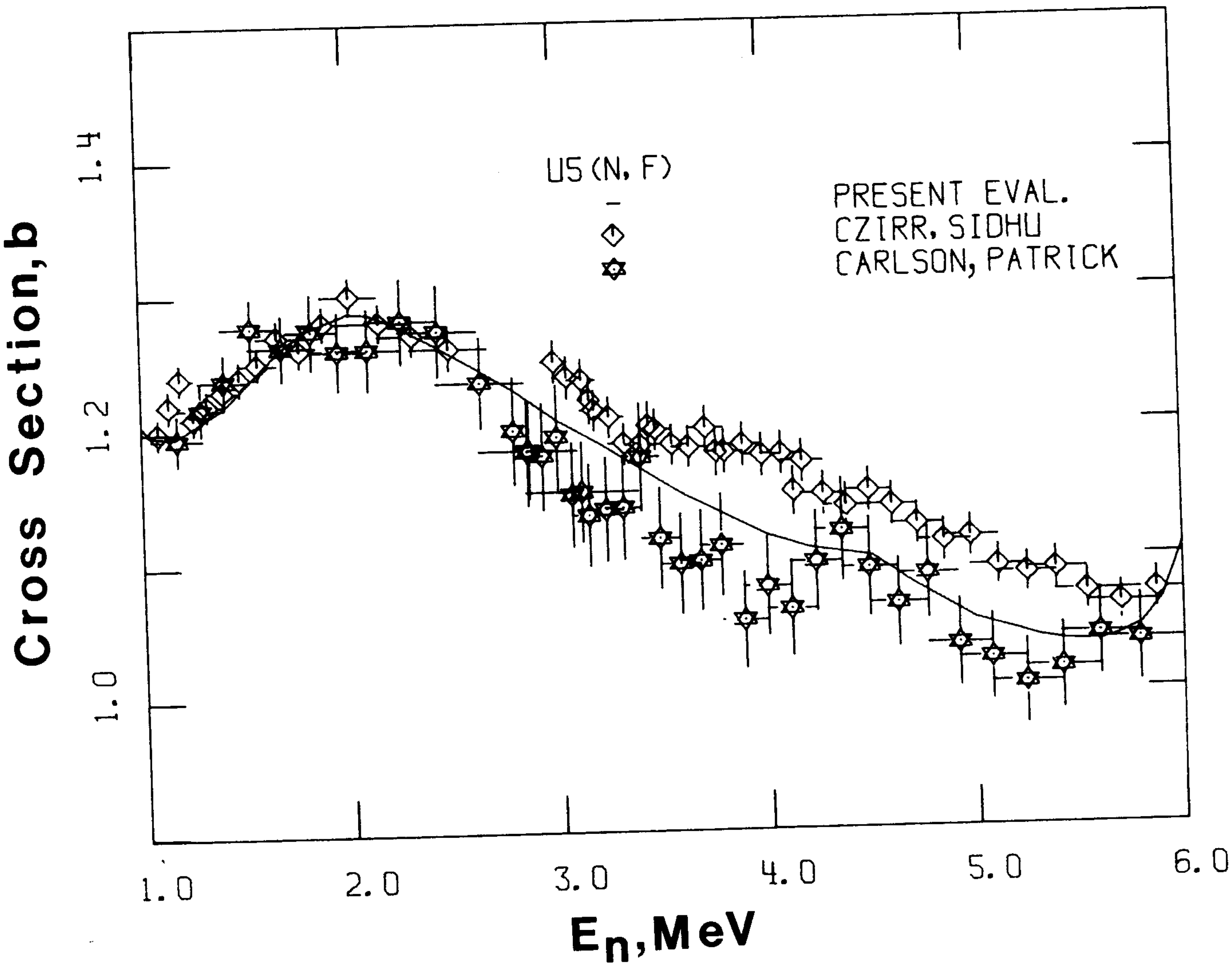

Fig. 17. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with Experimental Data.

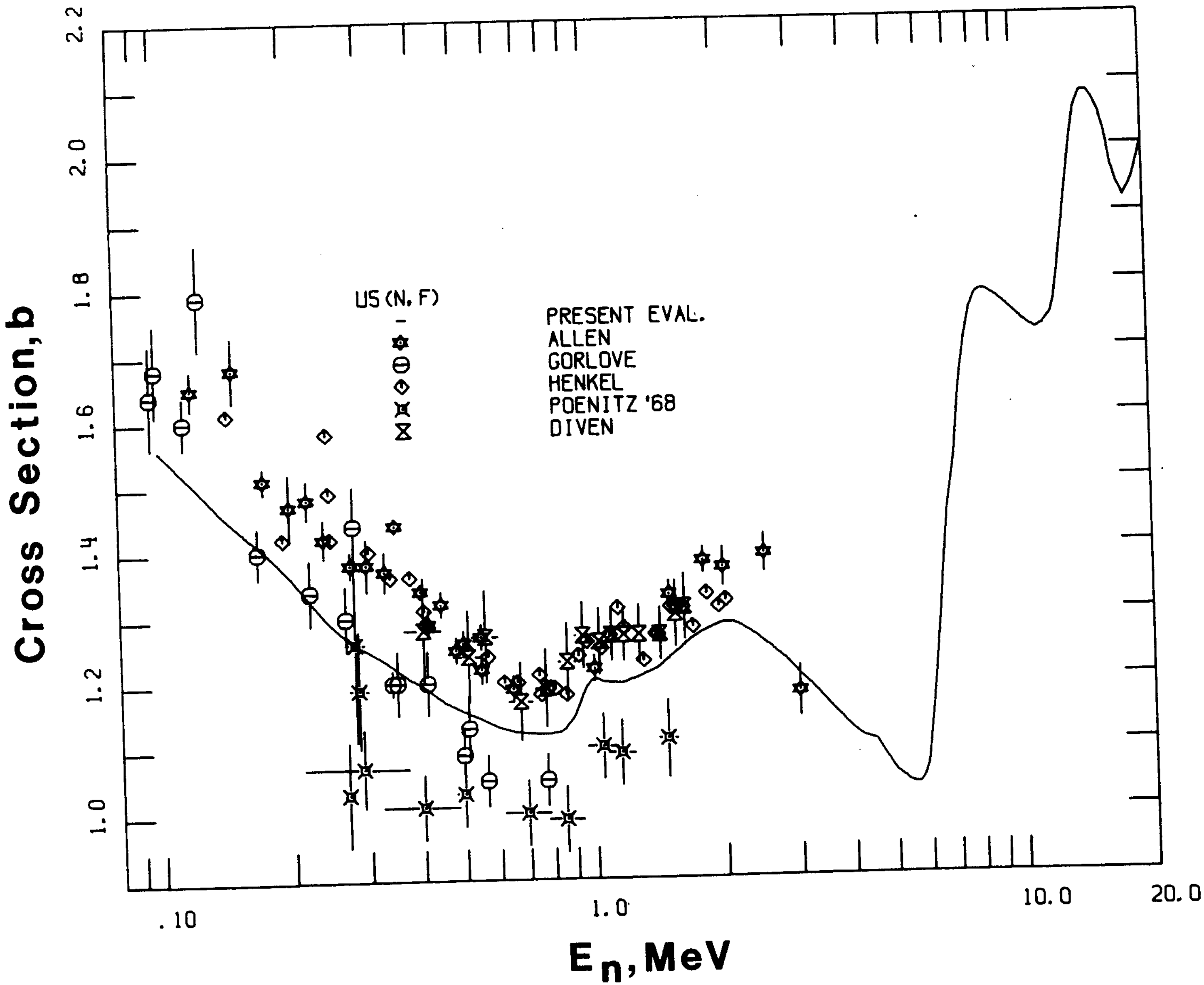

Fig. 18. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with Older Experimental Data Which were not Included in the Evaluation.

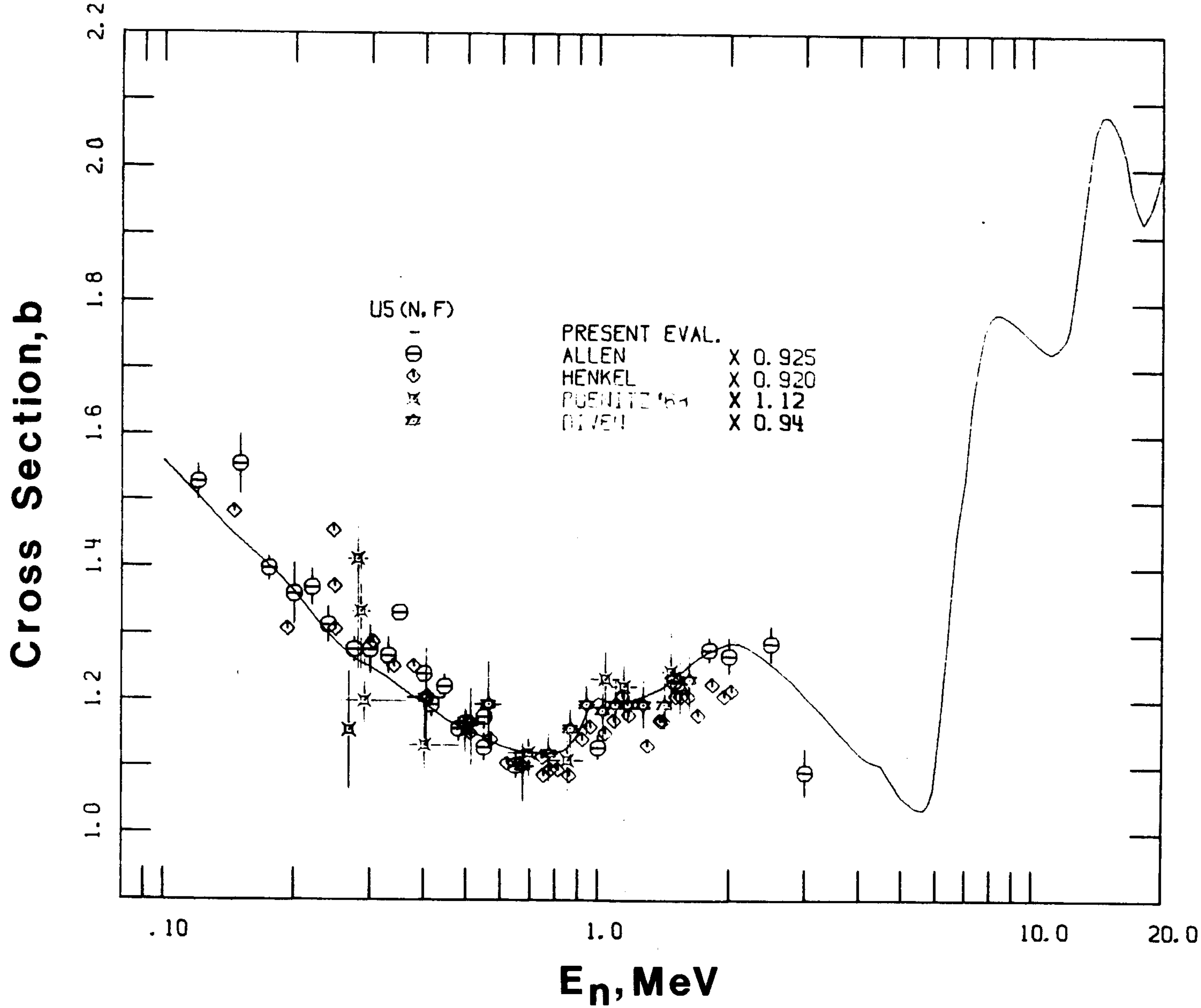

Fig. 19. Comparison of the Same Data as Shown in Fig. 18 but Normalized to the Present Result.

Table V. Results of the Present Evaluation

E_n, MeV	σ, b	E_n, MeV	σ, b
0.100	1.556	3.200	1.186
0.120	1.503	3.600	1.148
0.140	1.457	4.000	1.117
0.150	1.439	4.200	1.107
0.160	1.423	4.500	1.100
0.180	1.392	4.700	1.079
0.200	1.361	5.000	1.052
0.220	1.327	5.200	1.043
0.240	1.298	5.300	1.038
0.250	1.286	5.400	1.035
0.260	1.276	5.500	1.034
0.280	1.258	5.600	1.034
0.300	1.248	5.700	1.039
0.325	1.235	5.800	1.046
0.350	1.219	5.900	1.065
0.375	1.205	6.000	1.105
0.400	1.194	6.200	1.196
0.425	1.181	6.400	1.290
0.450	1.169	6.500	1.348
0.475	1.160	6.700	1.447
0.500	1.154	7.000	1.534
0.540	1.144	7.250	1.639
0.570	1.137	7.500	1.710
0.600	1.129	7.750	1.756
0.650	1.123	8.000	1.775
0.700	1.120	8.150	1.780
0.750	1.119	8.250	1.782
0.780	1.119	8.500	1.780
0.800	1.122	9 000	1.770
0.830	1.123	9.500	1.757
0.850	1.127	10.000	1.742
0.900	1.147	10.500	1.730
0.940	1.177	11.000	1.722
0.960	1.192	11.500	1.729
0.980	1.199	12.000	1.745
1.000	1.204	12.200	1.760
1.050	1.197	12.500	1.811
1.150	1.195	13.000	1.901
1.250	1.204	13.500	1.982
1.400	1.218	14.000	2.052
1.600	1.249	14.500	2.077
1.700	1.263	15.000	2.077
1.800	1.274	15.500	2.065
1.900	1.280	16.000	2.044
2.000	1.286	16.650	2.012
2.100	1.285	17.000	1.962
2.400	1.263	17.500	1.936
2.600	1.245	18.000	1.916
2.800	1.226	19.000	1.944
3.000	1.204	20.000	2.000

$^{235}U(n,f)$ cross section. The values masured by NBS[50] and the University of Michigan are considered very significant because a neutron source calibrated in an international source mesurement intercomparison was used in both experiments.

The average over the ^{252}Cf spontaneous fission neutron spectrum can be considered an integral over the differential cross section. However, the experiments used to measure the average cross section are very similar in design to the differential cross section experiments and the results were used here in the evaluation of the normalization of the differential cross section. The next logical step to test the evaluated $^{235}U(n,f)$ cross section would be calculations of the parameters of the benchmark critical reactor test facilities. However, such calculations are sensitive to uncertainties of other data. The only benchmark facility considered here is GODIVA which consists essentially of ^{235}U and thus depends on fewer other data than other critical benchmarks.

The results of recent calculations[53] of k_{eff} and reaction ratios for GODIVA with ENDF/B-V data indicate that the calculated spectrum is too hard.[54] This was concluded from the reaction ratios $\langle ^{28}C/^{28}f \rangle$ which was underpredicted by ~12% and $\langle ^{28}f/^{25}f \rangle$ which was overpredicted by ~6%. The differential data for the latter ratio are very well known. Therefore, $\langle ^{28}f/^{25}f \rangle$ was used to adjust the spectrum of GODIVA by changing the inelastic scattering of ^{235}U by ~15%. k_{eff} and the reaction ratios were then calculated using sensitivity coefficients.[55] The results are for the presently evaluated $^{235}U(n,f)$ cross section:

k_{eff}	0.999
$\langle ^{28}f/^{25}f \rangle$	1.003
$\langle ^{28}c/^{25}f \rangle$	0.983

V. CONCLUSIONS AND RECOMMENDATIONS

The result of the evaluation of the $^{235}U(n,f)$ cross section by an objective evaluation technique is very well supported by the experimental data which were reported with the lowest uncertainties (Arlt et al.,[35] Barton et al.,[40] Knoll,[51] Cance and Grenier, and Heaton et al.,[50]) and is in good agreement with many other data sets (Szabo,[44] Poenitz,[42] Smith et al.). Agreement with data by Wasson is reasonable after renormalization of these data with an evaluated resonance integral between 7.8 eV and 11.0 eV of 243.7 beV. Some of the more recently reported data are somewhat lower than the evaluation results (Poenitz,[41] Wasson and Meier[45]) and some are higher (Adamov et al.,[52] Kari[39]).

The shape of the cross section is very well established below 700 keV. The step in the cross section between 800 keV and 1 MeV is about 8%, however, the exact nature of this step is not well known. Several experiments, unfortunately, exclude the range of this step or do not cover a wide enough range. This results in some uncertainty of the above 8% which is considered a very important value.

> "It is recommended to measure the shape with good statistics and resolution as well as high point density between ~600 keV and 1.4 MeV."

The plateau in the cross section shape between 1.0 - 1.5 MeV is well established, however, the rise between 1.5 and 2.0 MeV is somewhat uncertain. Data by Szabo[44] and Grey Neutron Detector data by Poenitz[42] show a larger increase than the Black Neutron Detector data by Poenitz[44] and recent measurements relative to the H(n,n) cross section by Czirr and Sidhu, Kari[39] and Carlson and Patrick.[46] The data by Barton et al.[40] show a very flat shape in this range. It is believed that the shape is well-enough established by the majority of the data in this range.

There is a shape discrepancy between the plateau range (1 - 1.5 MeV) and the dip at ~5-6 MeV. there are two groups of data, the higher group[38,39,40] appears to predict a convex shape and the lower group [41,42,44,46] predicts a concave shape. The data by Barton et al. seem to "step-up" around 3 MeV, and the data by Carlson and Patrick show a peak at ~4.5 MeV. The dip appears to be at lower energy and much wider in the data by Kari, Carlson and Patrick, Szabo, and Poenitz than in the data by Barton et al. and Czirr and Sidhu.

> "Measurements of the shape between the plateau at 1.0 - 1.5 MeV and 6 MeV are recommended to resolve the shape discrepancy in this range."

The shape between 6 MeV and 14 MeV is reasonably well established, but large uncertainties exist above 14 MeV.

The normalization of the ^{235}U cross section appears to be well established by a large number of experiments. The latest reported data seem to increase the dispersion. However, the uncertainties of these data are such that they cannot suggest values differing from the present evaluation result one way or another.

> "It is recommended that future measurements be made with an uncertainty of ~1% in order to be significant."

APPENDIX I. The Data File

Smith, Henkel, Nobles (Set 1)

The present data source is a listing distributed by L. Stewart at a CSEWG meeting at Brookhaven National Laboratory in September 12, 1978. According to this distribution the cross sections were recalculated in 1975 by G. Hansen using values found in old data books. This listing replaces a previous reanalysis from 1968 in which the major changes from the original values were due to multiple scattering corrections. The new data listing quotes only statistical uncertainties which differ in some points by more than a factor 3 from the statistical uncertainties quoted in the 1968 distribution. The total uncertainty was recalculated from the 1968 quoted total uncertainties replacing the statistical uncertainty with the present (1975) quoted value(s) by

SQRT(E**2 - D**2), where E,D,S refer to quantities given in the 1968 listing.

Reference: R. K. Smith et al., Bull. Am. Phys. Soc., 2, 196 (K4), 5704.

See Figs. 4 and 11.

```
 U5(N,F) SET  1/128          SMITH ET AL,
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 .3000E 01 .0000E 00 .0000E 00 .1192E 01 .7830E-01 .7152E-02
 .4000E 01 .0000E 00 .0000E 00 .1128E 01 .6960E-01 .1466E-01
 .5000E 01 .0000E 00 .0000E 00 .1083E 01 .6250E-01 .1624E-01
 .5460E 01 .0000E 00 .0000E 00 .1083E 01 .6250E-01 .7581E-02
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 .8980E 01 .0000E 00 .0000E 00 .1770E 01 .9430E-01 .1770E-01
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 .1220E 02 .0000E 00 .0000E 00 .1714E 01 .1005E 00 .4285E-01
 .1320E 02 .0000E 00 .0000E 00 .1908E 01 .1165E 00 .5915E-01
 .1480E 02 .0000E 00 .0000E 00 .2046E 01 .1066E 00 .2046E-01
 .1600E 02 .0000E 00 .0000E 00 .2138E 01 .1205E 00 .5773E-01
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 .1900E 02 .0000E 00 .0000E 00 .2008E 01 .1077E 00 .2610E-01
 .1700E 22 .0000E 00 .0000E 00 .2154E 01 .1138E 00 .2800E-01
```

Barton et al. (Set 2)

The data source is the NSE publication. Energy scale errors have been suggested for these measurements, and it was pointed out that such energy error would cause substantial cross section changes due to the energy dependence of the reference H(n,n) cross section. The energy errors as suggested appear unlikely and the question was debated at the 1976 ANL meeting. Any change of the data seems unjustified. The energy uncertainty was not given in the paper and only a limit for the resolution $\geq 6\%$ was reported in the paper, therefore a resolution of 6% was assumed at all energies. The energy uncertainty was assumed to be 20% of the resolution (see Section III.2).

This data set is the one with the least quoted uncertainty (~1.5%), however, the uncertainty of the reference cross section due to energy uncertainties adds considerably to this value.

Reference: D. M. Barton et al., Nucl. Sci. Eng., 60, 369, 1976.

See Figs. 5 and 16.

U5(N,F) SET 2/128 BARTØN ET AL.

DATA

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.3000E 01	.0000E 00	.1800E 00	.1201E 01	.1254E-01	.3607E-02
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.3600E 01	.0000E 00	.2160E 00	.1173E 01	.1832E-01	.1407E-01
.3700E 01	.0000E 00	.2220E 00	.1147E 01	.1705E-01	.1262E-01
.3800E 01	.0000E 00	.2280E 00	.1156E 01	.1896E-01	.1503E-01
.4000E 01	.0000E 00	.2400E 00	.1129E 01	.1317E-01	.6781E-02
.4200E 01	.0000E 00	.2520E 00	.1135E 01	.2021E-01	.1589E-01
.4400E 01	.0000E 00	.2640E 00	.1110E 01	.1807E-01	.1332E-01
.4600E 01	.0000E 00	.2760E 00	.1090E 01	.1856E-01	.1417E-01
.4800E 01	.0000E 00	.2880E 00	.1100E 01	.1791E-01	.1320E-01
.5000E 01	.0000E 00	.3000E 00	.1081E 01	.1682E-01	.1190E-01

Barton et al. (Set 2) (Continued)

.5100E 01	.0000E 00	.3060E 00	.1077E 01	.1615E-01	.9685E-02
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.5600E 01	.0000E 00	.3360E 00	.1041E 01	.1626E-01	.1041E-01
.5700E 01	.0000E 00	.3420E 00	.1060E 01	.1726E-01	.1167E-01
.5800E 01	.0000E 00	.3480E 00	.1084E 01	.2622E-01	.2277E-01
.5900E 01	.0000E 00	.3540E 00	.1113E 01	.2138E-01	.1669E-01
.6000E 01	.0000E 00	.3600E 00	.1137E 01	.2460E-01	.2047E-01

Czirr and Sidhu (Set 31)

Two sets of measurements of the shape were made overlapping at 3.5 MeV. The low energy set has a gap due to experimental problems which partly extended also to the remaining points of this set around 3.5 MeV.

The statistical uncertainty of the overlap range of the two sets was added as a systematic uncertainty to the high energy set (0.9%).

The data by Czirr and Sidhu are quoted with a very low uncertainty and thus strongly influence the shape. This results in the only major problem of the shape evaluation where a discrepancy in the shape below 1 MeV is created (see Fig. 2). Breaking up the data set in the overlap range (~3.5 MeV) does not change the evaluated shape significantly. The data source was the Tables in the NSE publications.

Reference: J. B. Czirr and G. S. Sidhu, <u>Nucl. Sci. Eng.</u>, <u>60</u>, 383, 1976.

See Figs. 9, 16 and 17.

U5(N,F) SET 33/128 CZIRR,SIDHU I+II

DATA

.7540E 00	.5000E-03	.3100E-01	.1080E 01	.1200E-01	.1080E-01
.7860E 00	.0000E 00	.3250E-01	.1100E 01	.1200E-01	.1100E-01
.8190E 00	.0000E 00	.3450E-01	.1090E 01	.1000E-01	.8720E-02
.8550E 00	.0000E 00	.3750E-01	.1120E 01	.1100E-01	.8960E-02
.8940E 00	.0000E 00	.4000E-01	.1160E 01	.1100E-01	.9280E-02
.9350E 00	.0000E 00	.4250E-01	.1170E 01	.1100E-01	.9360E-02
.9790E 00	.0000E 00	.4550E-01	.1210E 01	.1100E-01	.9680E-02
.1026E 01	.7400E-03	.4900E-01	.1220E 01	.1200E-01	.9760E-02
.1077E 01	.0000E 00	.5300E-01	.1200E 01	.1100E-01	.9600E-02
.1132E 01	.0000E 00	.5700E-01	.1220E 01	.1200E-01	.9760E-02
.1191E 01	.0000E 00	.6100E-01	.1240E 01	.1100E-01	.9920E-02
.1254E 01	.0000E 00	.6600E-01	.1210E 01	.1100E-01	.9680E-02
.1323E 01	.0000E 00	.7200E-01	.1220E 01	.1100E-01	.9760E-02
.1398E 01	.0000E 00	.7800E-01	.1230E 01	.1100E-01	.9840E-02
.1479E 01	.0000E 00	.8500E-01	.1240E 01	.1300E-01	.9920E-02
.1568E 01	.0000E 00	.9300E-01	.1250E 01	.1100E-01	.1250E-01
.1665E 01	.0000E 00	.1015E 00	.1270E 01	.1300E-01	.1016E-01
.1771E 01	.0000E 00	.1110E 00	.1260E 01	.1200E-01	.1134E-01
.1887E 01	.0000E 00	.1220E 00	.1280E 01	.1300E-01	.1152E-01
.2015E 01	.2000E-02	.1350E 00	.1300E 01	.1400E-01	.1170E-01
.2157E 01	.0000E 00	.1500E 00	.1280E 01	.1400E-01	.1280E-01
.2315E 01	.0000E 00	.1565E 00	.1270E 01	.1500E-01	.1270E-01
.2490E 01	.0000E 00	.1655E 00	.1260E 01	.1600E-01	.1386E-01
.2990E 01	.0000E 00	.6300E-01	.1249E 01	.1573E-01	.1124E-01
.3050E 01	.4000E-02	.6500E-01	.1237E 01	.1565E-01	.1113E-01
.3120E 01	.0000E 00	.6500E-01	.1235E 01	.1564E-01	.1111E-01
.3156E 01	.0000E 00	.0000E 00	.1220E 01	.1800E-01	.1464E-01
.3180E 01	.0000E 00	.6500E-01	.1212E 01	.1549E-01	.1091E-01
.3250E 01	.0000E 00	.7000E-01	.1207E 01	.1546E-01	.1086E-01
.3320E 01	.0000E 00	.7500E-01	.1187E 01	.1533E-01	.1063E-01
.3400E 01	.0000E 00	.7500E-01	.1166E 01	.1533E-01	.1067E-01
.3438E 01	.0000E 00	.0000E 00	.1200E 01	.1500E-01	.1680E-01

Czirr and Sidhu (Set 31) (Continued)

.3470E 01	.0000E 00	.7500E-01	.1197E 01	.1540E-01	.1077E-01
.3550E 01	.0000E 00	.8000E-01	.1186E 01	.1533E-01	.1067E-01
.3630E 01	.0000E 00	.8000E-01	.1164E 01	.1532E-01	.1066E-01
.3710E 01	.0000E 00	.8500E-01	.1198E 01	.1613E-01	.1078E-01
.3759E 01	.0000E 00	.0000E 00	.1160E 01	.1800E-01	.1770E-01
.3800E 01	.0000E 00	.9000E-01	.1180E 01	.1602E-01	.1180E-01
.3890E 01	.0000E 00	.9000E-01	.1185E 01	.1605E-01	.1185E-01
.3980E 01	.6000E-02	.9000E-01	.1177E 01	.1601E-01	.1177E-01
.4070E 01	.0000E 00	.9500E-01	.1178E 01	.1601E-01	.1178E-01
.4129E 01	.0000E 00	.0000E 00	.1150E 01	.1800E-01	.1725E-01
.4170E 01	.0000E 00	.1000E 00	.1172E 01	.1598E-01	.1172E-01
.4270E 01	.0000E 00	.1050E 00	.1148E 01	.1584E-01	.1148E-01
.4380E 01	.0000E 00	.1100E 00	.1139E 01	.1578E-01	.1139E-01
.4490E 01	.0000E 00	.1100E 00	.1150E 01	.1585E-01	.1150E-01
.4600E 01	.0000E 00	.1150E 00	.1139E 01	.1578E-01	.1139E-01
.4720E 01	.0000E 00	.1250E 00	.1125E 01	.1570E-01	.1125E-01
.4850E 01	.0000E 00	.1300E 00	.1112E 01	.1563E-01	.1112E-01
.4980E 01	.8000E-02	.1300E 00	.1115E 01	.1564E-01	.1115E-01
.5110E 01	.0000E 00	.1350E 00	.1093E 01	.1476E-01	.1093E-01
.5250E 01	.0000E 00	.1400E 00	.1087E 01	.1472E-01	.1087E-01
.5390E 01	.0000E 00	.1450E 00	.1090E 01	.1550E-01	.1090E-01
.5540E 01	.0000E 00	.1550E 00	.1073E 01	.1540E-01	.1073E-01
.5700E 01	.0000E 00	.1650E 00	.1064E 01	.1458E-01	.1064E-01
.5870E 01	.0000E 00	.1700E 00	.1073E 01	.1540E-01	.1073E-01
.6040E 01	.1200E-01	.1750E 00	.1140E 01	.1579E-01	.1140E-01
.6220E 01	.0000E 00	.1800E 00	.1230E 01	.1633E-01	.1107E-01
.6400E 01	.0000E 00	.1900E 00	.1338E 01	.1772E-01	.1204E-01
.6600E 01	.0000E 00	.2050E 00	.1439E 01	.1962E-01	.1295E-01
.6810E 01	.0000E 00	.2100E 00	.1540E 01	.2193E-01	.1386E-01
.7020E 01	.0000E 00	.2200E 00	.1607E 01	.2309E-01	.1446E-01
.7250E 01	.0000E 00	.2300E 00	.1692E 01	.2514E-01	.1523E-01
.7480E 01	.0000E 00	.2400E 00	.1757E 01	.2791E-01	.1757E-01
.7730E 01	.0000E 00	.2550E 00	.1795E 01	.2893E-01	.1795E-01
.7990E 01	.1700E-01	.2700E 00	.1828E 01	.2910E-01	.1828E-01
.8270E 01	.0000E 00	.2800E 00	.1833E 01	.3079E-01	.2016E-01
.8550E 01	.0000E 00	.2950E 00	.1826E 01	.3076E-01	.2009E-01
.8860E 01	.0000E 00	.3150E 00	.1817E 01	.3157E-01	.1999E-01
.9180E 01	.0000E 00	.3300E 00	.1799E 01	.3148E-01	.2159E-01
.9520E 01	.0000E 00	.3500E 00	.1762E 01	.3314E-01	.2317E-01
.9860E 01	.0000E 00	.3650E 00	.1823E 01	.3419E-01	.2370E-01
.1025E 02	.2400E-01	.3900E 00	.1815E 01	.3416E-01	.2359E-01
.1066E 02	.0000E 00	.4150E 00	.1760E 01	.3401E-01	.2314E-01
.1108E 02	.0000E 00	.4350E 00	.1769E 01	.3582E-01	.2505E-01
.1153E 02	.0000E 00	.4650E 00	.1759E 01	.3660E-01	.2638E-01
.1201E 02	.0000E 00	.4950E 00	.1764E 01	.3851E-01	.2854E-01
.1252E 02	.0000E 00	.5250E 00	.1839E 01	.3962E-01	.2942E-01
.1306E 02	.0000E 00	.5600E 00	.1922E 01	.4175E-01	.3267E-01
.1364E 02	.0000E 00	.6000E 00	.2023E 01	.4578E-01	.3641E-01
.1426E 02	.0000E 00	.6450E 00	.2064E 01	.4875E-01	.3960E-01
.1493E 02	.4300E-01	.6900E 00	.2087E 01	.5061E-01	.4174E-01
.1564E 02	.0000E 00	.7350E 00	.2096E 01	.8317E-01	.7755E-01

Czirr and Sidhu (Set 31) (Continued)

.1640E 02	.0000E 00	.7900E 00	.2001E 01	.8103E-01	.7604E-01
.1722E 02	.0000E 00	.8550E 00	.1962E 01	.8193E-01	.7652E-01
.1811E 02	.0000E 00	.9250E 00	.1950E 01	.7217E-01	.6630E-01
.1907E 02	.0000E 00	.9950E 00	.1976E 01	.8684E-01	.8110E-01
.2010E 02	.6700E-01	.1079E 01	.2048E 01	.1313E 00	.1270E 00

Kari (Set 4)

The data were reported in a KFK report and presented at the Harwell Conference on Neutron Physics and Applications. It is here assumed that these data are replacing previous shape data reported by Leugers et al. at the ANL Specialists Meeting on Fission Cross Sections in 1976. The data reported by Kari are absolute. However, a notable fact is that the fission counting efficiency in this experiment was only 75%. This data set differs substantially above 13 MeV from the previously reported data by Leugers et al.

Reference: K. Kari, Nuclear Research Center Karlsruhe Report, KFK 2673, 1978.

See Figs. 8, 10, 14 and 15.

U5(N,F) SET 4/128 KARI(LEUGERS 78)

DATA

.1000E 01	.0000E 00	.0000E 00	.1162E 01	.4344E-01	.1424E-01
.1050E 01	.0000E 00	.0000E 00	.1233E 01	.4530E-01	.1485E-01
.1110E 01	.0000E 00	.0000E 00	.1233E 01	.4531E-01	.1485E-01
.1170E 01	.0000E 00	.0000E 00	.1274E 01	.4680E-01	.1534E-01
.1230E 01	.0000E 00	.0000E 00	.1274E 01	.4682E-01	.1534E-01
.1310E 01	.0000E 00	.0000E 00	.1245E 01	.4573E-01	.1499E-01
.1380E 01	.0000E 00	.0000E 00	.1255E 01	.4611E-01	.1511E-01
.1470E 01	.0000E 00	.0000E 00	.1306E 01	.4798E-01	.1572E-01
.1570E 01	.0000E 00	.0000E 00	.1306E 01	.4800E-01	.1573E-01
.1670E 01	.0000E 00	.0000E 00	.1327E 01	.4875E-01	.1598E-01
.1780E 01	.0000E 00	.0000E 00	.1327E 01	.4877E-01	.1598E-01
.1880E 01	.0000E 00	.0000E 00	.1328E 01	.4879E-01	.1599E-01
.1940E 01	.0000E 00	.0000E 00	.1328E 01	.4738E-01	.1599E-01
.2000E 01	.0000E 00	.0000E 00	.1338E 01	.4917E-01	.1423E-01
.2070E 01	.0000E 00	.0000E 00	.1339E 01	.4633E-01	.1234E-01
.2140E 01	.0000E 00	.0000E 00	.1339E 01	.4491E-01	.1234E-01
.2210E 01	.0000E 00	.0000E 00	.1339E 01	.4347E-01	.1235E-01
.2290E 01	.0000E 00	.0000E 00	.1339E 01	.3912E-01	.1235E-01
.2370E 01	.0000E 00	.0000E 00	.1330E 01	.3863E-01	.1226E-01
.2460E 01	.0000E 00	.0000E 00	.1320E 01	.3854E-01	.1217E-01
.2550E 01	.0000E 00	.0000E 00	.1310E 01	.3825E-01	.1208E-01
.2640E 01	.0000E 00	.0000E 00	.1300E 01	.3797E-01	.1198E-01
.2750E 01	.0000E 00	.0000E 00	.1300E 01	.3797E-01	.1199E-01
.2850E 01	.0000E 00	.0000E 00	.1290E 01	.3768E-01	.1190E-01
.2950E 01	.0000E 00	.0000E 00	.1280E 01	.3739E-01	.1180E-01
.3020E 01	.0000E 00	.0000E 00	.1280E 01	.3740E-01	.1181E-01
.3090E 01	.0000E 00	.0000E 00	.1260E 01	.3681E-01	.1162E-01
.3170E 01	.0000E 00	.0000E 00	.1250E 01	.3652E-01	.1153E-01
.3250E 01	.0000E 00	.0000E 00	.1250E 01	.3652E-01	.1153E-01
.3330E 01	.0000E 00	.0000E 00	.1251E 01	.3652E-01	.1153E-01
.3410E 01	.0000E 00	.0000E 00	.1241E 01	.3623E-01	.1144E-01
.3500E 01	.0000E 00	.0000E 00	.1220E 01	.3565E-01	.1125E-01
.3600E 01	.0000E 00	.0000E 00	.1221E 01	.3698E-01	.1125E-01
.3690E 01	.0000E 00	.0000E 00	.1211E 01	.3799E-01	.1116E-01
.3790E 01	.0000E 00	.0000E 00	.1211E 01	.3799E-01	.1116E-01
.3900E 01	.0000E 00	.0000E 00	.1201E 01	.3898E-01	.1107E-01

Kari (Set 4) (Continued)

.4000E 01	.0000E 00	.0000E 00	.1201E 01	.4027E-01	.9377E-02
.4120E 01	.0000E 00	.0000E 00	.1190E 01	.3993E-01	.9298E-02
.4240E 01	.0000E 00	.0000E 00	.1191E 01	.3993E-01	.9298E-02
.4360E 01	.0000E 00	.0000E 00	.1191E 01	.3993E-01	.9298E-02
.4490E 01	.0000E 00	.0000E 00	.1160E 01	.3892E-01	.9062E-02
.4620E 01	.0000E 00	.0000E 00	.1150E 01	.3858E-01	.8983E-02
.4760E 01	.0000E 00	.0000E 00	.1130E 01	.3790E-01	.8825E-02
.4910E 01	.0000E 00	.0000E 00	.1100E 01	.3688E-01	.8588E-02
.5060E 01	.0000E 00	.0000E 00	.1079E 01	.3620E-01	.8430E-02
.5230E 01	.0000E 00	.0000E 00	.1099E 01	.3688E-01	.8587E-02
.5400E 01	.0000E 00	.0000E 00	.1079E 01	.3620E-01	.8429E-02
.5570E 01	.0000E 00	.0000E 00	.1079E 01	.3620E-01	.8428E-02
.5760E 01	.0000E 00	.0000E 00	.1099E 01	.3687E-01	.8585E-02
.5960E 01	.0000E 00	.0000E 00	.1180E 01	.3703E-01	.9214E-02
.6160E 01	.0000E 00	.0000E 00	.1260E 01	.4091E-01	.9087E-02
.6380E 01	.0000E 00	.0000E 00	.1381E 01	.4334E-01	.1878E-01
.6610E 01	.0000E 00	.0000E 00	.1492E 01	.4681E-01	.2029E-01
.6850E 01	.0000E 00	.0000E 00	.1582E 01	.4793E-01	.2152E-01
.7020E 01	.0000E 00	.0000E 00	.1622E 01	.4914E-01	.2206E-01
.7110E 01	.0000E 00	.0000E 00	.1642E 01	.4975E-01	.2233E-01
.7200E 01	.0000E 00	.0000E 00	.1662E 01	.5036E-01	.2261E-01
.7290E 01	.0000E 00	.0000E 00	.1692E 01	.5127E-01	.2301E-01
.7380E 01	.0000E 00	.0000E 00	.1722E 01	.5218E-01	.2342E-01
.7570E 01	.0000E 00	.0000E 00	.1762E 01	.5339E-01	.2397E-01
.7670E 01	.0000E 00	.0000E 00	.1782E 01	.5399E-01	.2424E-01
.7770E 01	.0000E 00	.0000E 00	.1792E 01	.5429E-01	.2437E-01
.7870E 01	.0000E 00	.0000E 00	.1822E 01	.5520E-01	.2478E-01
.7970E 01	.0000E 00	.0000E 00	.1822E 01	.5520E-01	.2478E-01
.8070E 01	.0000E 00	.0000E 00	.1822E 01	.5519E-01	.2478E-01
.8180E 01	.0000E 00	.0000E 00	.1831E 01	.5549E-01	.2491E-01
.8290E 01	.0000E 00	.0000E 00	.1831E 01	.5548E-01	.2491E-01
.8400E 01	.0000E 00	.0000E 00	.1821E 01	.5517E-01	.2477E-01
.8520E 01	.0000E 00	.0000E 00	.1831E 01	.5547E-01	.2490E-01
.8630E 01	.0000E 00	.0000E 00	.1831E 01	.5546E-01	.2490E-01
.8750E 01	.0000E 00	.0000E 00	.1820E 01	.5515E-01	.2476E-01
.8870E 01	.0000E 00	.0000E 00	.1820E 01	.5514E-01	.2475E-01
.9000E 01	.0000E 00	.0000E 00	.1820E 01	.5514E-01	.2475E-01
.9120E 01	.0000E 00	.0000E 00	.1809E 01	.5482E-01	.2461E-01
.9250E 01	.0000E 00	.0000E 00	.1809E 01	.5481E-01	.2461E-01
.9380E 01	.0000E 00	.0000E 00	.1809E 01	.5481E-01	.2460E-01
.9520E 01	.0000E 00	.0000E 00	.1799E 01	.5449E-01	.2446E-01
.9660E 01	.0000E 00	.0000E 00	.1798E 01	.5448E-01	.2446E-01
.9800E 01	.0000E 00	.0000E 00	.1788E 01	.5417E-01	.2432E-01
.9940E 01	.0000E 00	.0000E 00	.1788E 01	.5416E-01	.2431E-01
.1009E 02	.0000E 00	.0000E 00	.1787E 01	.5415E-01	.2431E-01
.1024E 02	.0000E 00	.0000E 00	.1777E 01	.5384E-01	.2417E-01
.1039E 02	.0000E 00	.0000E 00	.1777E 01	.5383E-01	.2416E-01
.1055E 02	.0000E 00	.0000E 00	.1766E 01	.5351E-01	.2402E-01
.1071E 02	.0000E 00	.0000E 00	.1766E 01	.5350E-01	.2402E-01
.1087E 02	.0000E 00	.0000E 00	.1766E 01	.5349E-01	.2401E-01
.1104E 02	.0000E 00	.0000E 00	.1765E 01	.5348E-01	.2401E-01

Kari (Set 4) (Continued)

.1121E 02	.0000E 00	.0000E 00	.1765E 01	.5347E-01	.2400E-01
.1139E 02	.0000E 00	.0000E 00	.1774E 01	.5376E-01	.2414E-01
.1157E 02	.0000E 00	.0000E 00	.1784E 01	.5406E-01	.2427E-01
.1175E 02	.0000E 00	.0000E 00	.1784E 01	.5404E-01	.2426E-01
.1194E 02	.0000E 00	.0000E 00	.1803E 01	.5464E-01	.2453E-01
.1214E 02	.0000E 00	.0000E 00	.1823E 01	.5523E-01	.2479E-01
.1234E 02	.0000E 00	.0000E 00	.1853E 01	.5613E-01	.2520E-01
.1254E 02	.0000E 00	.0000E 00	.1892E 01	.5733E-01	.2574E-01
.1275E 02	.0000E 00	.0000E 00	.1932E 01	.5853E-01	.2627E-01
.1296E 02	.0000E 00	.0000E 00	.1971E 01	.5972E-01	.2681E-01
.1318E 02	.0000E 00	.0000E 00	.2011E 01	.6092E-01	.2735E-01
.1341E 02	.0000E 00	.0000E 00	.2050E 01	.6212E-01	.2788E-01
.1364E 02	.0000E 00	.0000E 00	.2090E 01	.6331E-01	.2842E-01
.1387E 02	.0000E 00	.0000E 00	.2119E 01	.6420E-01	.2882E-01
.1412E 02	.0000E 00	.0000E 00	.2148E 01	.6509E-01	.2922E-01
.1437E 02	.0000E 00	.0000E 00	.2158E 01	.6537E-01	.2935E-01
.1463E 02	.0000E 00	.0000E 00	.2177E 01	.6596E-01	.2961E-01
.1489E 02	.0000E 00	.0000E 00	.2176E 01	.6594E-01	.2960E-01
.1516E 02	.0000E 00	.0000E 00	.2176E 01	.6592E-01	.2959E-01
.1544E 02	.0000E 00	.0000E 00	.2175E 01	.6826E-01	.2958E-01
.1572E 02	.0000E 00	.0000E 00	.2144E 01	.7422E-01	.2917E-01
.1602E 02	.0000E 00	.0000E 00	.2124E 01	.7123E-01	.2888E-01
.1632E 02	.0000E 00	.0000E 00	.2093E 01	.7690E-01	.2847E-01
.1663E 02	.0000E 00	.0000E 00	.2072E 01	.7394E-01	.2819E-01
.1695E 02	.0000E 00	.0000E 00	.2042E 01	.7718E-01	.2777E-01
.1728E 02	.0000E 00	.0000E 00	.2011E 01	.7615E-01	.2488E-01
.1762E 02	.0000E 00	.0000E 00	.1981E 01	.7696E-01	.2450E-01
.1797E 02	.0000E 00	.0000E 00	.1940E 01	.7539E-01	.2400E-01
.1833E 02	.0000E 00	.0000E 00	.1909E 01	.7420E-01	.2362E-01
.1871E 02	.0000E 00	.0000E 00	.1889E 01	.7340E-01	.2336E-01
.1909E 02	.0000E 00	.0000E 00	.1888E 01	.7337E-01	.2335E-01
.1949E 02	.0000E 00	.0000E 00	.1897E 01	.7373E-01	.2347E-01
.1989E 02	.0000E 00	.0000E 00	.1916E 01	.7447E-01	.2370E-01
.2031E 02	.0000E 00	.0000E 00	.1926E 01	.7483E-01	.2382E-01

Szabo et al.

The present data source is the listing of final values in Szabo's paper presented at the Specialists' Meeting on Fission - Cross Sections in 1976 at ANL. The measurements reported at 1970 ANL were made with the fission chamber of White, therefore, this set is only used as shape data because their normalization is not independent from White's values. This also applied for one subset of data presented at the Conference on Neutron Physics - 1973 in Kiev and the new data presented at Argonne in 1976. All other data were made with independent fission chambers. An obvious bias between the results using White's chamber and the other chambers cannot be seen.

Reference: I. Szabo and J. P. Marquette, NEANDC/NEACRP Specialists' Meeting on Fast Neutron Fission Cross Sections, Argonne National Laboratory, ANL-76-90, 208 (1976).

See Figs. 6 and 16.

U5(N,F) SET 51/126 SZABO I

DATA

.1750E-01	.0000E 00	.3500E-02	.2150E 01	.9000E-01	.0000E 00
.1900E-01	.0000E 00	.3600E-02	.2480E 01	.9000E-01	.0000E 00
.2700E-01	.0000E 00	.3500E-02	.2100E 01	.8000E-01	.0000E 00
.4000E-01	.0000E 00	.3000E-02	.2047E 01	.7000E-01	.0000E 00
.4200E-01	.0000E 00	.5000E-02	.1800E 01	.6000E-01	.0000E 00
.5500E-01	.0000E 00	.3600E-02	.1822E 01	.5000E-01	.0000E 00
.6800E-01	.0000E 00	.5600E-02	.1765E 01	.4500E-01	.0000E 00
.7250E-01	.0000E 00	.6500E-02	.1740E 01	.5500E-01	.0000E 00
.7500E-01	.0000E 00	.3000E-02	.1767E 01	.5000E-01	.0000E 00
.8500E-01	.0000E 00	.4000E-02	.1556E 01	.4000E-01	.0000E 00
.9500E-01	.0000E 00	.5000E-02	.1540E 01	.5500E-01	.0000E 00
.1100E 00	.0000E 00	.1000E-01	.1530E 01	.5000E-01	.0000E 00
.1200E 00	.0000E 00	.6000E-02	.1570E 01	.5500E-01	.0000E 00
.1250E 00	.0000E 00	.7000E-02	.1500E 01	.5000E-01	.0000E 00
.1450E 00	.0000E 00	.9000E-02	.1500E 01	.5500E-01	.0000E 00
.1500E 00	.0000E 00	.6000E-02	.1450E 01	.4500E-01	.0000E 00
.1520E 00	.0000E 00	.1000E-01	.1440E 01	.4000E-01	.0000E 00
.1540E 00	.0000E 00	.1400E-01	.1440E 01	.3500E-01	.0000E 00
.1560E 00	.0000E 00	.1200E-01	.1450E 01	.4500E-01	.0000E 00
.1950E 00	.0000E 00	.1100E-01	.1365E 01	.5500E-01	.0000E 00
.2150E 00	.0000E 00	.1000E-01	.1325E 01	.4500E-01	.0000E 00
.2270E 00	.0000E 00	.1600E-01	.1295E 01	.3500E-01	.0000E 00
.2510E 00	.0000E 00	.1100E-01	.1285E 01	.3500E-01	.0000E 00
.2570E 00	.0000E 00	.1500E-01	.1275E 01	.5500E-01	.0000E 00
.2720E 00	.0000E 00	.1500E-01	.1275E 01	.4500E-01	.0000E 00
.2860E 00	.0000E 00	.1500E-01	.1270E 01	.3500E-01	.0000E 00
.3130E 00	.0000E 00	.1500E-01	.1265E 01	.4500E-01	.0000E 00
.3200E 00	.0000E 00	.8000E-02	.1190E 01	.4500E-01	.0000E 00
.3310E 00	.0000E 00	.1500E-01	.1210E 01	.4500E-01	.0000E 00
.3690E 00	.0000E 00	.1500E-01	.1215E 01	.4500E-01	.0000E 00
.4070E 00	.0000E 00	.1500E-01	.1205E 01	.3500E-01	.0000E 00
.5060E 00	.0000E 00	.1700E-01	.1160E 01	.3000E-01	.0000E 00
.5400E 00	.0000E 00	.1000E-01	.1160E 01	.4500E-01	.0000E 00
.6550E 00	.0000E 00	.2200E-01	.1140E 01	.3500E-01	.0000E 00

Szabo et al. (Continued)

.7300E 00	.0000E 00	.3000E-01	.1140E 01	.3000E-01	.0000E 00
.8100E 00	.0000E 00	.3500E-01	.1135E 01	.3500E-01	.0000E 00
.1010E 01	.0000E 00	.4000E-01	.1205E 01	.3500E-01	.0000E 00
.1020E 01	.0000E 00	.2500E-01	.1187E 01	.3500E-01	.0000E 00
.1080E 01	.0000E 00	.2500E-01	.1167E 01	.3500E-01	.0000E 00
.1280E 01	.0000E 00	.2400E-01	.1207E 01	.3500E-01	.0000E 00
.1405E 01	.0000E 00	.2300E-01	.1229E 01	.3500E-01	.0000E 00
.1485E 01	.0000E 00	.2200E-01	.1255E 01	.3000E-01	.0000E 00
.1530E 01	.0000E 00	.2200E-01	.1252E 01	.3500E-01	.0000E 00
.1680E 01	.0000E 00	.2100E-01	.1272E 01	.3500E-01	.0000E 00
.1800E 01	.0000E 00	.2000E-01	.1306E 01	.3500E-01	.0000E 00
.1915E 01	.0000E 00	.2000E-01	.1353E 01	.3500E-01	.0000E 00
.2000E 01	.0000E 00	.1900E-01	.1315E 01	.3000E-01	.0000E 00
.2100E 01	.0000E 00	.1700E-01	.1316E 01	.3500E-01	.0000E 00

.1150E-01	.0000E 00	.3000E-02	.2760E 01	.9000E-01	.0000E 00
.1500E-01	.0000E 00	.3000E-02	.2500E 01	.7000E-01	.0000E 00
.1700E-01	.0000E 00	.3000E-02	.2370E 01	.8000E-01	.0000E 00
.1700E-01	.0000E 00	.3000E-02	.2420E 01	.8000E-01	.0000E 00
.2250E-01	.0000E 00	.2500E-02	.2200E 01	.6000E-01	.0000E 00
.3300E-01	.0000E 00	.5000E-02	.2020E 01	.6000E-01	.0000E 00
.3600E-01	.0000E 00	.3000E-02	.1984E 01	.6500E-01	.0000E 00
.3500E-01	.0000E 00	.3000E-02	.1975E 01	.6000E-01	.0000E 00
.4600E-01	.0000E 00	.5000E-02	.1850E 01	.5000E-01	.0000E 00
.5100E-01	.0000E 00	.3000E-02	.1863E 01	.5500E-01	.0000E 00
.5100E-01	.0000E 00	.3000E-02	.1849E 01	.5000E-01	.0000E 00
.5500E-01	.0000E 00	.3000E-02	.1830E 01	.5000E-01	.0000E 00
.7100E-01	.0000E 00	.3000E-02	.1650E 01	.4500E-01	.0000E 00
.7100E-01	.0000E 00	.3000E-02	.1710E 01	.4500E-01	.0000E 00
.7600E-01	.0000E 00	.2500E-02	.1700E 01	.5000E-01	.0000E 00
.8350E-01	.0000E 00	.1100E-01	.1650E 01	.5000E-01	.0000E 00
.9300E-01	.0000E 00	.4000E-02	.1550E 01	.4000E-01	.0000E 00
.1035E 00	.0000E 00	.5500E-02	.1530E 01	.4000E-01	.0000E 00
.1160E 00	.0000E 00	.1500E-01	.1520E 01	.4000E-01	.0000E 00
.1240E 00	.0000E 00	.4000E-02	.1540E 01	.4000E-01	.0000E 00
.1240E 00	.0000E 00	.4000E-02	.1580E 01	.4000E-01	.0000E 00
.1350E 00	.0000E 00	.5000E-02	.1420E 01	.4000E-01	.0000E 00
.1500E 00	.0000E 00	.5000E-02	.1460E 01	.4000E-01	.0000E 00
.1720E 00	.0000E 00	.5500E-02	.1460E 01	.4000E-01	.0000E 00
.1990E 00	.0000E 00	.5500E-02	.1420E 01	.4000E-01	.0000E 00
.8600E 00	.0000E 00	.2600E-01	.1140E 01	.3500E-01	.0000E 00
.9200E 00	.0000E 00	.2600E-01	.1185E 01	.3500E-01	.0000E 00
.2040E 01	.0000E 00	.1900E-01	.1330E 01	.3500E-01	.0000E 00
.2180E 01	.0000E 00	.1800E-01	.1294E 01	.3500E-01	.0000E 00
.2190E 01	.0000E 00	.1700E-01	.1303E 01	.3000E-01	.0000E 00
.2280E 01	.0000E 00	.1700E-01	.1304E 01	.3000E-01	.0000E 00
.2300E 01	.0000E 00	.1700E-01	.1293E 01	.3000E-01	.0000E 00
.2350E 01	.0000E 00	.3200E-01	.1256E 01	.4000E-01	.0000E 00
.2380E 01	.0000E 00	.1800E-01	.1275E 01	.3200E-01	.0000E 00
.2590E 01	.0000E 00	.3100E-01	.1219E 01	.4000E-01	.0000E 00
.2610E 01	.0000E 00	.1600E-01	.1270E 01	.3000E-01	.0000E 00
.2780E 01	.0000E 00	.3000E-01	.1206E 01	.4000E-01	.0000E 00

Szabo et al. (Continued)

.2650E 01	.0000E 00	.3000E-01	.1203E 01	.3600E-01	.0000E 00
.3090E 01	.0000E 00	.2900E-01	.1167E 01	.3500E-01	.0000E 00
.3230E 01	.0000E 00	.2800E-01	.1156E 01	.3400E-01	.0000E 00
.3360E 01	.0000E 00	.2000E-01	.1130E 01	.3300E-01	.0000E 00
.3550E 01	.0000E 00	.2500E-01	.1137E 01	.3500E-01	.0000E 00
.3800E 01	.0000E 00	.2400E-01	.1100E 01	.3500E-01	.0000E 00
.3920E 01	.0000E 00	.1010E 00	.1088E 01	.3800E-01	.0000E 00
.4470E 01	.0000E 00	.6300E-01	.1100E 01	.4000E-01	.0000E 00
.5015E 01	.0000E 00	.3700E-01	.1000E 01	.3600E-01	.0000E 00
.5530E 01	.0000E 00	.2400E-01	.1030E 01	.3600E-01	.0000E 00

White (Set 6)

Present data source is the publication in J. Nucl. Energy. ENDF/B-V values for H(n,n) were substituted for all energies below 5 MeV. This caused changes of a few tenths of one percent for most values but 1 percent at one energy. The 5.4 MeV value is based on a recalculation by G. Hansen (memorandum by L. Stewart, August 24, 1978) of a correction factor of 1.0199 for useing ENDF/B-V cross section data for the H(n,n) reference and a relativistic correction.

A case could be made to renormalize these data upward by approximately 1% based on the $T_{1/2}$ measurement of U-234 by White and the presently known value of ~$2.447 \cdot 10^4$ y; however, the relationship of the foils used in the $T_{1/2}$ - determination and the cross sections measurements is not exactly known. Szabo confirms that the mass derived with the high $T_{1/2}$ - value is within 1% of White's values.

Reference: P. H. White, J. Nucl. Energy A/B, 19, 325, 1965.

See Figs. 4, 10, 14 and 15.

```
  U5(N,F) SET  6/128        WHITE
 DATA
  .4000E-01 .0000E 00 .4000E-02 .2103E 01 .6300E-01 .0000E 00
  .6700E-01 .0000E 00 .5000E-02 .1768E 01 .5304E-01 .0000E 00
  .1270E 00 .0000E 00 .1500E-01 .1536E 01 .3850E-01 .0000E 00
  .1600E 00 .0000E 00 .2000E-01 .1519E 01 .3600E-01 .0000E 00
  .2070E 00 .0000E 00 .2000E-01 .1376E 01 .3450E-01 .0000E 00
  .3120E 00 .0000E 00 .2000E-01 .1289E 01 .3250E-01 .0000E 00
  .4040E 00 .0000E 00 .2000E-01 .1222E 01 .3050E-01 .0000E 00
  .5050E 00 .0000E 00 .2000E-01 .1147E 01 .2875E-01 .0000E 00
  .5050E 00 .0000E 00 .2000E-01 .1177E 01 .2950E-01 .0000E 00
  .1000E 01 .0000E 00 .5000E-01 .1221E 01 .3050E-01 .0000E 00
  .2250E 01 .0000E 00 .5000E-01 .1333E 01 .3960E-01 .0000E 00
  .5400E 01 .0000E 00 .1000E 00 .1020E 01 .5000E-01 .0000E 00
  .1410E 02 .0000E 00 .5000E-01 .2170E 01 .4340E-01 .0000E 00
```

Poenitz, BND (Set 71)

Measurements with a Black Neutron Detector (BND) were reported in 1974 and the measurements were extended to higher and lower energies in 1977. The measurements in 1974 were shape measurements with two absolute values at 800 keV and at 3.5 MeV. All values reported in 1977 were absolute. The measurement at 3.5 MeV was made with a sample provided by EURATOM, the measurements at 800 keV and those reported in 1977 were made with different samples provided by Meadows. The latter data set has been revised for more recent determinations of the mass scale of the samples involved (1.3%). The 800 keV value was measured with another sample by Meadows and is unchanged.

Poenitz, GND (Set 72)

This is a set of data measured with a Grey Neutron Detector (GND), which was reported in 1974. These are only shape data.

Poenitz, AA (Set 73)

This set contains absolute values obtained with the associated activity technique. Values were reported in 1974.

Poenitz, VSO_4

This set contains an absolute value measured with a VSO_4 - Bath.

References: W. P. Poenitz, Nucl. Sci. Eng., 53, 370, 1974, Nucl. Sci. Eng., 60, 894, 1977.

See Figs. 7, 8, 11, 13 and 15.

U5(N,F) SET 71 PØENITZ I
DATA

.3500 E 01	.0000 E 00	.0000 E 00	.1198 E 01	.2600 E-01	.1677 E-01
.8000 E 00	.0000 E 00	.0000 E 00	.1134 E 01	.2700 E-01	.1361 E-01
.3990 E 00	.0000 E 00	.3000 E-01	.1223 E 01	.4400 E-01	.2691 E-01
.4990 E 00	.0000 E 00	.3000 E-01	.1164 E 01	.2100 E-01	.1397 E-01
.6000 E 00	.0000 E 00	.2300 E-01	.1107 E 01	.1300 E-01	.1107 E-01
.6850 E 00	.0000 E 00	.2700 E-01	.1136 E 01	.2000 E-01	.1136 E-01
.7990 E 00	.0000 E 00	.2700 E-01	.1134 E 01	.2100 E-01	.1134 E-01
.8470 E 00	.0000 E 00	.1200 E-01	.1115 E 01	.2200 E-01	.1226 E-01
.8670 E 00	.0000 E 00	.1200 E-01	.1156 E 01	.2400 E-01	.1387 E-01
.8870 E 00	.0000 E 00	.1200 E-01	.1147 E 01	.2500 E-01	.1491 E-01
.8970 E 00	.0000 E 00	.1200 E-01	.1157 E 01	.2700 E-01	.1735 E-01
.9170 E 00	.0000 E 00	.1200 E-01	.1174 E 01	.2400 E-01	.1409 E-01
.9470 E 00	.0000 E 00	.1200 E-01	.1220 E 01	.2700 E-01	.1464 E-01
.9660 E 00	.0000 E 00	.1200 E-01	.1217 E 01	.2800 E-01	.1460 E-01
.9970 E 00	.0000 E 00	.1200 E-01	.1194 E 01	.3000 E-01	.1433 E-01
.1017 E 01	.0000 E 00	.1200 E-01	.1190 E 01	.3300 E-01	.1785 E-01
.1047 E 01	.0000 E 00	.1200 E-01	.1170 E 01	.3000 E-01	.1404 E-01
.1100 E 01	.0000 E 00	.2500 E-01	.1200 E 01	.2600 E-01	.1440 E-01

Poenitz, BND (Set 71) (Continued)

.1185E 01	.0000E 00	.2300E-01	.1205E 01	.2900E-01	.1325E-01
.1401E 01	.0000E 00	.2300E-01	.1196E 01	.3000E-01	.1316E-01
.1491E 01	.0000E 00	.2300E-01	.1213E 01	.3050E-01	.1334E-01
.1550E 01	.0000E 00	.2300E-01	.1225E 01	.3200E-01	.1225E-01
.1601E 01	.0000E 00	.2200E-01	.1235E 01	.3200E-01	.1235E-01
.1650E 01	.0000E 00	.2200E-01	.1261E 01	.3300E-01	.1261E-01
.1701E 01	.0000E 00	.2200E-01	.1254E 01	.3250E-01	.1254E-01
.1750E 01	.0000E 00	.2200E-01	.1258E 01	.3300E-01	.1258E-01
.1801E 01	.0000E 00	.2100E-01	.1249E 01	.3250E-01	.1249E-01
.1850E 01	.0000E 00	.2100E-01	.1283E 01	.3300E-01	.1283E-01
.1903E 01	.0000E 00	.2100E-01	.1282E 01	.3300E-01	.1282E-01
.1951E 01	.0000E 00	.2100E-01	.1284E 01	.3300E-01	.1284E-01
.2000E 01	.0000E 00	.2100E-01	.1285E 01	.3300E-01	.1285E-01
.2012E 01	.0000E 00	.2000E-01	.1300E 01	.3200E-01	.1300E-01
.2053E 01	.0000E 00	.2000E-01	.1285E 01	.3300E-01	.1285E-01
.2093E 01	.0000E 00	.2000E-01	.1264E 01	.3250E-01	.1264E-01
.2133E 01	.0000E 00	.2000E-01	.1272E 01	.3300E-01	.1272E-01
.2202E 01	.0000E 00	.2000E-01	.1288E 01	.3350E-01	.1288E-01
.2253E 01	.0000E 00	.2000E-01	.1259E 01	.3400E-01	.1259E-01
.2302E 01	.0000E 00	.2000E-01	.1281E 01	.3400E-01	.1281E-01
.2403E 01	.0000E 00	.1900E-01	.1315E 01	.3200E-01	.1315E-01
.2703E 01	.0000E 00	.1800E-01	.1243E 01	.3200E-01	.1243E-01
.2803E 01	.0000E 00	.1800E-01	.1233E 01	.3100E-01	.1233E-01
.1930E 00	.1500E-01	.9000E-02	.1336E 01	.3652E-01	.2437E-01
.2130E 00	.1500E-01	.9000E-02	.1325E 01	.2961E-01	.1879E-01
.2150E 00	.5000E-02	.5000E-02	.1325E 01	.4935E-01	.1610E-01
.2250E 00	.5000E-02	.5000E-02	.1294E 01	.3751E-01	.1311E-01
.2330E 00	.1500E-01	.9000E-02	.1287E 01	.2566E-01	.1565E-01
.2350E 00	.5000E-02	.5000E-02	.1292E 01	.3356E-01	.1309E-01
.2450E 00	.5000E-02	.5000E-02	.1280E 01	.2862E-01	.1297E-01
.2530E 00	.1500E-01	.9000E-02	.1280E 01	.2566E-01	.1556E-01
.2550E 00	.5000E-02	.5000E-02	.1277E 01	.2665E-01	.1294E-01
.2650E 00	.5000E-02	.5000E-02	.1266E 01	.2665E-01	.1283E-01
.2710E 00	.1500E-01	.1300E-01	.1245E 01	.2270E-01	.1513E-01
.2750E 00	.5000E-02	.5000E-02	.1209E 01	.2566E-01	.1225E-01
.2850E 00	.5000E-02	.5000E-02	.1230E 01	.2665E-01	.1246E-01
.2930E 00	.1500E-01	.8000E-02	.1238E 01	.2764E-01	.1881E-01
.2950E 00	.5000E-02	.5000E-02	.1241E 01	.2665E-01	.1131E-01
.3050E 00	.5000E-02	.1000E-01	.1239E 01	.2566E-01	.1004E-01
.6840E 00	.1600E-01	.1500E-01	.1123E 01	.2270E-01	.1138E-01
.7350E 00	.1600E-01	.1500E-01	.1129E 01	.2270E-01	.1258E-01
.7930E 00	.1600E-01	.1500E-01	.1104E 01	.2171E-01	.1231E-01
.8140E 00	.1600E-01	.1400E-01	.1111E 01	.2171E-01	.1351E-01
.8350E 00	.1600E-01	.1500E-01	.1136E 01	.2764E-01	.1957E-01
.8500E 00	.1600E-01	.1400E-01	.1120E 01	.2270E-01	.1248E-01
.8710E 00	.1600E-01	.1400E-01	.1136E 01	.2369E-01	.1381E-01
.8930E 00	.1600E-01	.1500E-01	.1140E 01	.2369E-01	.1386E-01
.9090E 00	.1600E-01	.1500E-01	.1139E 01	.2369E-01	.1385E-01
.9290E 00	.1600E-01	.1500E-01	.1143E 01	.3454E-01	.2895E-01
.9690E 00	.1600E-01	.1500E-01	.1198E 01	.2467E-01	.1457E-01
.9880E 00	.1600E-01	.1500E-01	.1187E 01	.2467E-01	.1564E-01

Poenitz, BND (Set 71) (Continued)

.1088E 01	.1600E-01	.1600E-01	.1185E 01	.3751E-01	.3243E-01
.1138E 01	.1600E-01	.1500E-01	.1193E 01	.2467E-01	.1451E-01
.1185E 01	.1600E-01	.1500E-01	.1193E 01	.2862E-01	.2055E-01
.1236E 01	.1600E-01	.1400E-01	.1183E 01	.2862E-01	.2038E-01
.1288E 01	.1600E-01	.1400E-01	.1205E 01	.2566E-01	.1465E-01
.1336E 01	.1600E-01	.1400E-01	.1201E 01	.2566E-01	.1460E-01
.1388E 01	.1600E-01	.1500E-01	.1175E 01	.2467E-01	.1428E-01
.1433E 01	.1600E-01	.1400E-01	.1209E 01	.2566E-01	.1470E-01
.1486E 01	.1600E-01	.1600E-01	.1192E 01	.2566E-01	.1450E-01
.1578E 01	.1700E-01	.1400E-01	.1231E 01	.2566E-01	.1496E-01
.1778E 01	.1700E-01	.1300E-01	.1229E 01	.2862E-01	.1743E-01
.1978E 01	.1800E-01	.1300E-01	.1246E 01	.2764E-01	.1514E-01
.2178E 01	.1800E-01	.1200E-01	.1246E 01	.2764E-01	.1514E-01
.2222E 01	.1800E-01	.1000E-01	.1269E 01	.2764E-01	.1543E-01
.2472E 01	.1900E-01	.1100E-01	.1208E 01	.2764E-01	.1469E-01
.2718E 01	.1900E-01	.9000E-02	.1185E 01	.2566E-01	.1321E-01
.2968E 01	.1000E-01	.1000E-01	.1135E 01	.2566E-01	.1495E-01
.3162E 01	.1000E-01	.9000E-02	.1119E 01	.2566E-01	.1588E-01
.3262E 01	.1000E-01	.8000E-02	.1122E 01	.2467E-01	.1592E-01
.3360E 01	.1000E-01	.8000E-02	.1119E 01	.2566E-01	.1588E-01
.3463E 01	.1100E-01	.9000E-02	.1118E 01	.2566E-01	.1586E-01
.3658E 01	.1100E-01	.8000E-02	.1079E 01	.2665E-01	.1749E-01
.3756E 01	.1100E-01	.8000E-02	.1082E 01	.2467E-01	.1534E-01
.3955E 01	.1200E-01	.7000E-02	.1082E 01	.2665E-01	.1534E-01
.4153E 01	.1200E-01	.7000E-02	.1055E 01	.2862E-01	.1603E-01
.4351E 01	.1300E-01	.6000E-02	.1054E 01	.2961E-01	.1602E-01
.4396E 01	.1300E-01	.1330E 00	.1069E 01	.2270E-01	.1083E-01
.4449E 01	.1300E-01	.6000E-02	.1051E 01	.3060E-01	.1597E-01
.4618E 01	.1300E-01	.1250E 00	.1061E 01	.2171E-01	.1075E-01
.5179E 01	.1400E-01	.1070E 00	.1015E 01	.2171E-01	.1028E-01
.5536E 01	.1500E-01	.4400E-01	.1013E 01	.2369E-01	.1231E-01
.5618E 01	.1500E-01	.1020E 00	.1020E 01	.2764E-01	.1549E-01
.5714E 01	.1600E-01	.8500E-01	.1035E 01	.2665E-01	.1364E-01
.5809E 01	.1600E-01	.7200E-01	.1019E 01	.2665E-01	.1238E-01
.5898E 01	.1600E-01	.6900E-01	.1059E 01	.2961E-01	.1609E-01
.6006E 01	.1700E-01	.7000E-01	.1073E 01	.3257E-01	.2065E-01
.6107E 01	.1700E-01	.3800E-01	.1139E 01	.3158E-01	.1731E-01
.6185E 01	.1700E-01	.6800E-01	.1146E 01	.3060E-01	.1393E-01
.6384E 01	.1700E-01	.6400E-01	.1244E 01	.3553E-01	.1890E-01
.6579E 01	.0000E 00	.6200E-01	.1369E 01	.2961E-01	.2080E-01
.6803E 01	.0000E 00	.2000E-02	.1495E 01	.3454E-01	.2272E-01
.7025E 01	.0000E 00	.4900E-01	.1521E 01	.3652E-01	.2466E-01
.7478E 01	.0000E 00	.4800E-01	.1691E 01	.5330E-01	.2912E-01
.7575E 01	.0000E 00	.4700E-01	.1772E 01	.5823E-01	.3231E-01
.8275E 01	.0000E 00	.6100E-01	.1770E 01	.6119E-01	.3227E-01

** 76 ** 76 ** 76 ** 76 ** 76 ** 76 ** 76 ** 76 ** 76 ** 76 ** 76 ** 76 ** 76 ** 76 **

Poenitz, BND (Set 71) (Continued)

```
U5(N,F) SET 72     PØENITZ II
DATA
 ,3500E-01  ,0000E 00  ,3000E-02  .2017E 01  ,7600E-01
 ,4700E-01  ,0000E 00  ,5000E-02  .1824E 01  ,6000E-01
 ,6800E-01  ,0000E 00  ,5000E-02  .1690E 01  ,4500E-01
 ,8400E-01  ,0000E 00  ,5000E-02  ,1579E 01  ,3800E-01
 ,9700E-01  ,0000E 00  ,8000E-02  ,1579E 01  ,3500E-01
 ,1210E 00  ,0000E 00  ,8000E-02  .1502E 01  ,3100E-01
 ,1425E 00  ,0000E 00  ,1900E-01  .1409E 01  ,2700E-01
 ,1765E 00  ,0000E 00  ,1750E-01  ,1370E 01  ,2300E-01
 ,1940E 00  ,0000E 00  ,2300E-01  ,1384E 01  ,4200E-01
 ,2180E 00  ,0000E 00  ,1800E-01  ,1348E 01  ,4600E-01
 ,2310E 00  ,0000E 00  ,2500E-01  ,1326E 01  ,1800E-01
 ,2730E 00  ,0000E 00  ,3500E-01  ,1358E 01  ,4000E-01
 ,2820E 00  ,0000E 00  ,2500E-01  .1290E 01  ,2000E-01
 ,2970E 00  ,0000E 00  ,1000E-01  ,1259E 01  ,2900E-01
 ,3960E 00  ,0000E 00  ,3000E-01  .1191E 01  ,2450E-01
 ,4305E 00  ,0000E 00  ,2200E-01  .1179E 01  ,3100E-01
 ,5083E 00  ,0000E 00  ,2433E-01  ,1133E 01  ,2200E-01
 ,5560E 00  ,0000E 00  ,2800E-01  ,1190E 01  ,1800E-01
 ,5985E 00  ,0000E 00  ,2750E-01  ,1139E 01  ,1650E-01
 ,6470E 00  ,0000E 00  .2700E-01  .1113E 01  ,1700E-01
 ,6920E 00  ,0000E 00  ,2500E-01  ,1104E 01  ,1800E-01
 ,7085E 00  ,0000E 00  ,2650E-01  .1103E 01  ,2400E-01
 ,7570E 00  ,0000E 00  ,2400E-01  .1119E 01  ,2000E-01
 ,8080E 00  ,0000E 00  ,2600E-01  ,1102E 01  ,2400E-01
 ,8610E 00  ,0000E 00  ,2300E-01  ,1158E 01  ,2700E-01
 ,8870E 00  ,0000E 00  ,2800E-01  .1175E 01  ,2600E-01
 ,9000E 00  ,0000E 00  ,2600E-01  .1143E 01  ,2100E-01
 ,9440E 00  ,0000E 00  ,4500E-01  ,1252E 01  ,2200E-01
 ,9585E 00  ,0000E 00  ,2500E-01  ,1228E 01  ,2950E-01
 ,9970E 00  ,0000E 00  ,2500E-01  .1212E 01  ,2400E-01
 ,1038E 01  ,0000E 00  ,4800E-01  ,1210E 01  ,2600E-01
 ,1106E 01  ,0000E 00  ,4350E-01  .1196E 01  ,2500E-01
 ,1196E 01  ,0000E 00  ,4450E-01  .1231E 01  ,2500E-01
 ,1227E 01  ,0000E 00  ,4200E-01  ,1239E 01  ,2700E-01
 ,1278E 01  ,0000E 00  ,4100E-01  ,1205E 01  ,2100E-01
 ,1300E 01  ,0000E 00  ,4300E-01  .1154E 01  ,2600E-01
 ,1410E 01  ,0000E 00  ,4450E-01  .1242E 01  ,2800E-01
 ,1421E 01  ,0000E 00  ,4000E-01  .1263E 01  ,2800E-01
 ,1506E 01  ,0000E 00  ,4000E-01  .1259E 01  ,2700E-01
 ,1605E 01  ,0000E 00  ,5500E-01  ,1299E 01  ,2900E-01
 ,1704E 01  ,0000E 00  ,7400E-01  ,1286E 01  ,2800E-01
 ,1820E 01  ,0000E 00  ,5067E-01  .1289E 01  ,3000E-01
 ,1911E 01  ,0000E 00  ,3300E-01  ,1294E 01  ,3300E-01
 ,1970E 01  ,0000E 00  ,7200E-01  ,1338E 01  ,3100E-01
 ,2015E 01  ,0000E 00  ,5033E-01  .1368E 01  ,3233E-01
 ,2107E 01  ,0000E 00  ,6800E-01  .1338E 01  ,3500E-01
 ,2204E 01  ,0000E 00  ,6700E-01  .1337E 01  ,3600E-01
 ,2304E 01  ,0000E 00  ,6600E-01  .1354E 01  ,3700E-01
```

Poenitz, BND (Set 71) (Continued)

```
.2499E 01 .0000E 00 .6400E-01 .1250E 01 .3600E-01
.2820E 01 .0000E 00 .4533E-01 .1294E 01 .3800E-01
.3000E 01 .0000E 00 .5800E-01 .1234E 01 .4000E-01
.3500E 01 .0000E 00 .5300E-01 .1192E 01 .4500E-01

**76**76**76**76**76**76**76**76**76**76**76**76**76**

U5(N,F) SET 73     PØENITZ III
DATA
.4480E 00 .0000E 00 .6000E-01 .1220E 01 .4400E-01
.5525E 00 .0000E 00 .5500E-01 .1115E 01 .4200E-01
.6010E 00 .0000E 00 .5550E-01 .1108E 01 .4050E-01
.6440E 00 .0000E 00 .6500E-01 .1101E 01 .4200E-01

**76**76**76**76**76**76**76**76**76**76**76**76**76**76

U5(N,F) SET 74     PØENITZ IV
DATA
.4980E+00 .0000E+00 .0000E+00 .1151E+01 .4200E-01

**76**76**76**76**76**76**76**76**76**76**76**76**76**76
```

Wasson (Set 10)

Data were reported at the 1976 ANL meeting normalized to a 7.8-11 eV integral reported by Deruytter. Normalization to the average of several measurements as given in Table I and reference to the thermal cross section of ENDF/B-V causes an upward normalization by 2.2%. The uncertainty of the normalized data consists of several components which were not added by the author. The various uncertainties were summarized by Poenitz.[5] The uncertainty for the 7.8-11 eV integral was substituted with the present value of Table I. This yields a value of 2.5% for the uncertainties of the normalization which is added to a typical 1.6-2.0% uncertainty of the quoted data. The data in the keV energy range were measured relative to the H(n,n) cross section and an analysis by Gammel was used for this cross section. The data were normalized here with the ENDF/B-V H(n,n) cross section.

Reference: O. A. Wasson, NEANDC/NEACRP Specialists' Meeting on the Fast Neutron Cross Sections, Argonne National Laboratory, ANL-76-90, 183, 1976.

See Figs. 1 and 8.

U5(N,F) SET 10/128 WASSEN

DATA

.5500E-02	.0000E 00	.5000E-03	.4055E 01	.3082E-02	.4022E-01
.6500E-02	.0000E 00	.5000E-03	.3345E 01	.1318E 00	.4021E-01
.7500E-02	.0000E 00	.5000E-03	.3305E 01	.1281E 00	.4021E-01
.8500E-02	.0000E 00	.5000E-03	.3010E 01	.1160E 00	.4020E-01
.9500E-02	.0000E 00	.5000E-03	.3213E 01	.1211E 00	.4020E-01
.1500E-01	.0000E 00	.5000E-02	.2532E 01	.8802E-01	.1004E-01
.2500E-01	.0000E 00	.5000E-02	.2157E 01	.7488E-01	.2006E-01
.3500E-01	.0000E 00	.5000E-02	.1981E 01	.7433E-01	.2005E-01
.4500E-01	.0000E 00	.5000E-02	.1816E 01	.6270E-01	.2003E-01
.5500E-01	.0000E 00	.5000E-02	.1766E 01	.6173E-01	.2002E-01
.6500E-01	.0000E 00	.5000E-02	.1730E 01	.5979E-01	.2001E-01
.7500E-01	.0000E 00	.5000E-02	.1644E 01	.5706E-01	.2000E-01
.8500E-01	.0000E 00	.5000E-02	.1602E 01	.5730E-01	.1999E-01
.9500E-01	.0000E 00	.5000E-02	.1604E 01	.5733E-01	.1998E-01
.1250E 00	.0000E 00	.2500E-01	.1465E 01	.4972E-01	.9984E-02
.1750E 00	.0000E 00	.2500E-01	.1391E 01	.4723E-01	.9977E-02
.2500E 00	.0000E 00	.5000E-01	.1251E 01	.4224E-01	.9974E-02
.3500E 00	.0000E 00	.5000E-01	.1192E 01	.4019E-01	.9976E-02
.4500E 00	.0000E 00	.5000E-01	.1161E 01	.4197E-01	.9982E-02
.5500E 00	.0000E 00	.5000E-01	.1123E 01	.4099E-01	.9989E-02
.6500E 00	.0000E 00	.5000E-01	.1115E 01	.4028E-01	.9997E-02
.7500E 00	.0000E 00	.5000E-01	.1100E 01	.3993E-01	.1000E-01

Kaeppeler (Sets 111 and 112)

Measurements were made relative to H(n,n). The data were revised to use ENDF/B-V for the reference cross section. Only the set 111 is absolute data.

Reference: F. Kaeppeler, Nuclear Research Center Karlsruhe report, KFK 1772, 1973.

```
U5(N,F) SET111     KAEPPELER I
DATA
 .5460E 00  .0000E 00  .2200E-01  .1205E 01  .4104E-01
 .6620E 00  .0000E 00  .2300E-01  .1213E 01  .3159E-01
 .7580E 00  .0000E 00  .2300E-01  .1163E 01  .3026E-01
 .9080E 00  .0000E 00  .2200E-01  .1193E 01  .3460E-01
 .1057E 01  .0000E 00  .2600E-01  .1249E 01  .3744E-01
 .1125E 01  .0000E 00  .2500E-01  .1258E 01  .4270E-01
 .1175E 01  .0000E 00  .2500E-01  .1223E 01  .4151E-01

     **76**76**76**76**76**76**76**76**76**76**76**76**76**76**

U5(N,F) SET112     KAEPPELER II
DATA
 .5130E 00  .0000E 00  .3200E-01  .1208E 01  .1089E-01
 .5600E 00  .0000E 00  .2000E-01  .1191E 01  .2147E-01
 .6780E 00  .0000E 00  .2100E-01  .1206E 01  .1450E-01
 .7670E 00  .0000E 00  .2000E-01  .1164E 01  .1048E-01
 .7950E 00  .0000E 00  .2000E-01  .1177E 01  .1414E-01
 .8720E 00  .0000E 00  .2000E-01  .1102E 01  .2314E-01
 .9200E 00  .0000E 00  .2200E-01  .1141E 01  .1711E-01
 .9300E 00  .0000E 00  .2000E-01  .1172E 01  .2227E-01
 .9450E 00  .0000E 00  .2400E-01  .1206E 01  .1327E-01
 .9660E 00  .0000E 00  .2100E-01  .1213E 01  .1456E-01
 .1013E 01  .0000E 00  .2000E-01  .1294E 01  .2457E-01
 .1060E 01  .0000E 00  .2200E-01  .1228E 01  .1472E-01
 .1164E 01  .0000E 00  .4000E-01  .1245E 01  .2239E-01

     **76**76**76**76**76**76**76**76**76**76**76**76**76**76**
```

U. Michigan (Set 91)

Values for several photoneutron sources were reported at various occasions. The current values are used in the evaluation.

Data are from a private communication with G. Knoll and represent values to be published in Annals of Nucl. Sci., September 1978).

```
 U5(N,F) SET 91/128        U. MICHIGAN I
DATA
 .1400E 00 .0000E 00 .0000E 00 .1471E 01 .2900E-01 .0000E 00
 .2650E 00 .0000E 00 .0000E 00 .1274E 01 .2600E-01 .0000E 00
 .7700E 00 .0000E 00 .0000E 00 .1162E 01 .2500E-01 .0000E 00
 .9640E 00 .0000E 00 .0000E 00 .1195E 01 .2600E-01 .0000E 00
```

U. Michigan II (Set 92)

An average value over the Cf spectrum was obtained with the same $MnSO_4$ bath and foils used in the measurements with the photoneutron sources.

Reference: G. F. Knoll, Proc. Symp. on Neutron Standards and Applications, NBS Special Publication 493, 3041, 1977.

See Fig. 5.

```
 U5(N,F) SET 92     U. MICHIGAN II
DATA
 .2200E+01 .0000E 00 .0000E 00 .1215E 01 .2200E-01

        ** 76 ** 76 ** 76 ** 76 ** 76 ** 76 ** 76 ** 76 ** 76 ** 76 ** 76 ** 76 ** 76 ** 76
```

Carlson and Patrick (Set 151)

The shape of the cross section was measured relative to the H(n,n) cross section. The measurements were reported at the Harwell Conference on Neutron Physics and Applications and the data are still preliminary. Measurements were made up to 20 MeV but data only up to 6 MeV were provided for the present evaluation. The data consists of two sets which overlap around 3 MeV. Combining the two data sets requires the addition of 2.2% for the statistical uncertainty of the overlap range to the uncertainty of either data set. The 2.2% were here added to the uncertainty of the data above 3 MeV.

Reference: A. D. Carlson and B. H. Patrick, Conference on Neutron Physics and Applications, Harwell, England (1978).

See Figs. 12 and 17.

U5(N,F) SET153/128 CARLSON,PATRICK
DATA

.1171E 01	.2281E-01	.1250E 00	.1195E 01	.1872E-01	.1700E-01
.1288E 01	.2631E-01	.1085E 00	.1217E 01	.1941E-01	.1700E-01
.1404E 01	.2994E-01	.1230E 00	.1238E 01	.2055E-01	.1900E-01
.1535E 01	.3423E-01	.1405E 00	.1277E 01	.2185E-01	.2000E-01
.1686E 01	.3941E-01	.1625E 00	.1262E 01	.2439E-01	.2200E-01
.1830E 01	.4456E-01	.1220E 00	.1274E 01	.2828E-01	.2600E-01
.1956E 01	.4932E-01	.1350E 00	.1259E 01	.2910E-01	.2800E-01
.2100E 01	.5478E-01	.1505E 00	.1260E 01	.3034E-01	.2900E-01
.2259E 01	.6111E-01	.1675E 00	.1281E 01	.3201E-01	.3100E-01
.2406E 01	.6644E-01	.1875E 00	.1273E 01	.3421E-01	.3300E-01
.2635E 01	.7699E-01	.2115E 00	.1234E 01	.3387E-01	.3300E-01
.2788E 01	.0000E 00	.0000E 00	.1198E 01	.3969E-01	.2800E-01
.2855E 01	.0000E 00	.0000E 00	.1182E 01	.3945E-01	.2800E-01
.2860E 01	.8795E-01	.2380E 00	.1183E 01	.3531E-01	.3400E-01
.2924E 01	.0000E 00	.0000E 00	.1179E 01	.3940E-01	.2700E-01
.2995E 01	.0000E 00	.0000E 00	.1193E 01	.3889E-01	.2700E-01
.3069E 01	.0000E 00	.0000E 00	.1150E 01	.3822E-01	.2700E-01
.3114E 01	.9891E-01	.2720E 00	.1152E 01	.3662E-01	.3600E-01
.3147E 01	.1005E-01	.7868E-01	.1135E 01	.3800E-01	.2700E-01
.3227E 01	.1044E-01	.6172E-01	.1139E 01	.3838E-01	.2800E-01
.3311E 01	.1034E-01	.8491E-01	.1140E 01	.3852E-01	.2800E-01
.3397E 01	.1127E-01	.8526E-01	.1177E 01	.3983E-01	.2900E-01
.3487E 01	.1172E-01	.9180E-01	.1117E 01	.3772E-01	.2700E-01
.3581E 01	.1220E-01	.9553E-01	.1098E 01	.3717E-01	.2700E-01
.3678E 01	.1270E-01	.9946E-01	.1100E 01	.3725E-01	.2700E-01
.3780E 01	.1323E-01	.1036E 00	.1111E 01	.3768E-01	.2700E-01
.3885E 01	.1377E-01	.1080E 00	.1055E 01	.3588E-01	.2600E-01
.3996E 01	.1433E-01	.1126E 00	.1080E 01	.3649E-01	.2700E-01
.4111E 01	.1500E-01	.1176E 00	.1063E 01	.3589E-01	.2600E-01
.4231E 01	.1555E-01	.1228E 00	.1096E 01	.3724E-01	.2700E-01
.4356E 01	.1637E-01	.1283E 00	.1121E 01	.3611E-01	.2800E-01
.4487E 01	.1711E-01	.1341E 00	.1093E 01	.3712E-01	.2700E-01
.4625E 01	.1790E-01	.1403E 00	.1066E 01	.3344E-01	.2700E-01
.4768E 01	.1874E-01	.1469E 00	.1088E 01	.3396E-01	.2300E-01
.4918E 01	.1963E-01	.1540E 00	.1035E 01	.3260E-01	.2200E-01

Carlson and Patrick (Set 151) (Continued)

.5076E 01	.2059E-01	.1515E 00	.1024E 01	.3221E-01	.2200E-01
.5241E 01	.2160E-01	.1694E 00	.1005E 01	.3169E-01	.2200E-01
.5415E 01	.2268E-01	.1779E 00	.1016E 01	.3207E-01	.2200E-01
.5597E 01	.2384E-01	.1870E 00	.1041E 01	.3287E-01	.2200E-01
.5789E 01	.2507E-01	.1968E 00	.1036E 01	.3280E-01	.2200E-01
.5991E 01	.2639E-01	.2072E 00	.1082E 01	.3385E-01	.2300E-01
.6203E 01	.2781E-01	.2183E 00	.1158E 01	.3591E-01	.2400E-01

Cancé and Grenier (Set 18)

An absolute value at 14.6 MeV was obtained with the associated particle technique and reported at the NEANDC/NEACRP - Specialists Meeting on Fast Neutron Fission Cross Sections. Additional values at 13.9 MeV and at 2.5 MeV were reported at the Conference on Neutron Physics and Applications at Harwell. The neutron flux was compared at 2.5 MeV with the directional counter by Szabo and at 14 MeV with a Black Neutron Detector.

Reference: M. Cancé and G. Grenier, Nucl. Sci. and Eng., 68, 197 (1978).

See Figs. 7 and 10.

```
  U5(N,F) SET 18/128          CANCE
 DATA
 .2500E 01 .0000E 00 .0000E 00 .1261E 01 .0000E 00 .0000E 00
 .1390E 02 .0000E 00 .0000E 00 .2062E 01 .3900E-01 .0000E 00
 .1460E 02 .0000E 00 .0000E 00 .2063E 01 .3900E-01 .0000E 00
```

Wasson and Meier (Set 17)

Absolute cross section data were obtained with a Black Neutron Detector. The mass of the ^{235}U in the fission chamber is based upon a comparison with a standard NBS foil set and is therefore not independent from the measurement by Heaton et al., (Set 13). The data are preliminary and some additional values were measured recently but are not included in the present evaluation.

Reference: Private communication by A. O. Wasson, National Bureau of Standards, 1978.

See Fig. 5.

U5(N,F) SET 17/128 WASSON,MEIER

DATA

.2540E	00	.5000E-02	.2750E-01	.1300E	01	.5100E-01	.2080E-01
.2700E	00	.5000E-02	.2500E-01	.1233E	01	.4800E-01	.1849E-01
.3180E	00	.5000E-02	.2150E-01	.1205E	01	.4600E-01	.1807E-01
.3550E	00	.5000E-02	.2200E-01	.1197E	01	.4500E-01	.1676E-01
.3950E	00	.5000E-02	.1900E-01	.1211E	01	.5400E-01	.1574E-01
.4350E	00	.5000E-02	.2350E-01	.1156E	01	.4300E-01	.1387E-01
.4780E	00	.5000E-02	.1900E-01	.1128E	01	.4100E-01	.1241E-01
.5230E	00	.5000E-02	.2450E-01	.1118E	01	.3900E-01	.7826E-02
.5520E	00	.5000E-02	.2450E-01	.1117E	01	.3900E-01	.8936E-02
.5950E	00	.5000E-02	.2500E-01	.1099E	01	.3900E-01	.9891E-02
.6370E	00	.5000E-02	.2600E-01	.1107E	01	.3900E-01	.1107E-01
.6840E	00	.5000E-02	.2350E-01	.1098E	01	.3900E-01	.1098E-01
.7190E	00	.6000E-02	.2700E-01	.1091E	01	.3800E-01	.7637E-02
.7630E	00	.6000E-02	.2600E-01	.1102E	01	.3900E-01	.1102E-01
.8010E	00	.6000E-02	.3100E-01	.1097E	01	.3900E-01	.1097E-01
.8450E	00	.6000E-02	.2950E-01	.1127E	01	.4000E-01	.1127E-01
.8830E	00	.6000E-02	.2950E-01	.1091E	01	.4000E-01	.1091E-01
.9220E	00	.5000E-02	.5700E-01	.1132E	01	.4100E-01	.1132E-01
.9630E	00	.1000E-01	.4350E-01	.1151E	01	.4300E-01	.1381E-01
.1010E	01	.5000E-02	.2250E-01	.1174E	01	.4400E-01	.1057E-01
.1021E	01	.5000E-02	.2100E-01	.1207E	01	.4700E-01	.1328E-01
.1065E	01	.5000E-02	.2250E-01	.1179E	01	.4300E-01	.1179E-01
.1071E	01	.5000E-02	.1900E-01	.1168E	01	.4400E-01	.1285E-01
.1119E	01	.5000E-02	.2050E-01	.1159E	01	.4300E-01	.1159E-01
.1167E	01	.5000E-02	.2200E-01	.1163E	01	.4300E-01	.1163E-01
.1217E	01	.5000E-02	.2250E-01	.1157E	01	.4300E-01	.1157E-01

Gayther (Set 8)

The shape of the cross section was measured with a Grey Neutron Detector. The fission events were measured by detecting the fast fission neutrons, thus the actual measured quantity was $\nu \cdot \sigma_f$.

Reference: D. B. Gayther, Proc. of Conference on Nuclear Cross Sections and Technology, NBS Sp. Publ. 425, Vol. II, 564 (1975).

See Fig. 12.

U5(N,F) SET 8/128 GAYTHER
DATA

.1500E-02	.0000E 00	.5000E-03	.7861E 01	.3231E 00	.0000E 00
.2500E-02	.0000E 00	.5000E-03	.5722E 01	.2232E 00	.0000E 00
.3500E-02	.0000E 00	.5000E-03	.5045E 01	.1867E 00	.0000E 00
.4500E-02	.0000E 00	.5000E-03	.4474E 01	.1566E 00	.0000E 00
.5500E-02	.0000E 00	.5000E-03	.4046E 01	.1336E 00	.0000E 00
.6500E-02	.0000E 00	.5000E-03	.3379E 01	.1081E 00	.0000E 00
.7500E-02	.0000E 00	.5000E-03	.3280E 01	.1017E 00	.0000E 00
.8500E-02	.0000E 00	.5000E-03	.3071E 01	.9520E-01	.0000E 00
.9500E-02	.0000E 00	.5000E-03	.3153E 01	.9774E-01	.0000E 00
.1250E-01	.0000E 00	.2500E-02	.2702E 01	.8100E-01	.0000E 00
.1750E-01	.0000E 00	.2500E-02	.2359E 01	.7100E-01	.0000E 00
.2250E-01	.0000E 00	.2500E-02	.2237E 01	.6700E-01	.0000E 00
.2750E-01	.0000E 00	.2500E-02	.2095E 01	.6300E-01	.0000E 00
.3500E-01	.0000E 00	.5000E-02	.1978E 01	.5934E-01	.0000E 00
.4500E-01	.0000E 00	.5000E-02	.1893E 01	.5679E-01	.0000E 00
.5500E-01	.0000E 00	.5000E-02	.1866E 01	.5598E-01	.0000E 00
.6500E-01	.0000E 00	.5000E-02	.1815E 01	.5445E-01	.0000E 00
.7500E-01	.0000E 00	.5000E-02	.1733E 01	.5372E-01	.0000E 00
.8500E-01	.0000E 00	.5000E-02	.1612E 01	.5158E-01	.0000E 00
.9500E-01	.0000E 00	.5000E-02	.1568E 01	.5240E-01	.0000E 00
.1250E 00	.0000E 00	.2500E-01	.1505E 01	.5100E-01	.0000E 00
.1750E 00	.0000E 00	.2500E-01	.1363E 01	.4700E-01	.0000E 00
.2250E 00	.0000E 00	.2500E-01	.1306E 01	.4400E-01	.0000E 00
.2750E 00	.0000E 00	.2500E-01	.1227E 01	.4200E-01	.0000E 00
.3500E 00	.0000E 00	.5000E-01	.1196E 01	.4066E-01	.0000E 00
.4500E 00	.0000E 00	.5000E-01	.1153E 01	.4035E-01	.0000E 00
.5500E 00	.0000E 00	.5000E-01	.1134E 01	.4082E-01	.0000E 00
.6500E 00	.0000E 00	.5000E-01	.1122E 01	.4264E-01	.0000E 00
.7500E 00	.0000E 00	.5000E-01	.1122E 01	.4376E-01	.0000E 00
.8500E 00	.0000E 00	.5000E-01	.1156E 01	.4508E-01	.0000E 00
.9500E 00	.0000E 00	.5000E-01	.1218E 01	.4872E-01	.0000E 00

Kuks et al. (Set 12)

A value was obtained at 2.5 MeV using the associated particle technique.

Reference: I. M. Kuks et al., Conf. on Neutron Physics, Kiev, Vol. 4, 18, (1973).

See Fig. 4.

```
U5(N,F) SET 12/128        KUKS ET AL.
DATA
.2500E 01 .0000E 00 .0000E 00 .1310E 01 .5000E-01 .0000E 00
```

Alkharov et al. (Set 16)

A value was obtained at 14.8 MeV with the associated particle technique.

Reference: See V. M. Adamov et al., Proc. of a Symposium Neutron Standards and Applications, NBS Spec. Publ. 493, 313, (1977).

```
U5(N,F) SET 16/128        ALKHAROV ET AL.
DATA
.1480E 02 .0000E 00 .0000E 00 .2188E 01 .3700E-01 .0000E 00
```

Arlt (Set 251)

An absolute value was measured at 14.7 MeV using the associated particle technique.

Reference: R. Arlt et al., International Nuclear Data Committee report INDC/GDR/-7.16, 10, 1978.

See Figs.: 5 and 9

```
U5(N,F) SET251/128          ARLT DRESDEN
DATA
.1470E 02 .0000E 00 .0000E 00 .2073E 01 .2360E-01 .0000E 00
```

Adamov et al. (Set 14)

An average of the cross section over the ^{252}Cf fission neutron spectrum was measured. The technique was based upon a coincidence technique.

Reference: V. M. Adamov et al., Proceedings of a Symposium on Neutron Standards and Applications, NBS Spec. Publ. 493, 313, (1977).

```
U5(N,F) SET 14     ADAMØV
.2100E 01 .0000E 00 .0000E 00 .1265E 01 .1900E-01
```

7676**76**76**76**76**76**76**76**76**76**76**76**76

APPENDIX II: The Evaluation of $^{235}U(n,f)$ above 100 KeV for ENDF/B-V

The $^{235}U(n,f)$ cross section of ENDF/B-V above 100 KeV was derived from a preliminary version of the present evaluation. Small differences in the shape are due to the later discovery that two data sets[28,39] used the analysis of the H(n,n) cross section by Gammel[29] instead of ENDF/B-V[16,17] values. However, the major difference is in normalization. The members of the Subcommittee on Normalizations and Standards of the Cross Section Evaluation Working Group (CSEWG) selected normalization factors closely corresponding to the first column of Table III, but the value by Arlt _et al._[35] was excluded because it was not well documented at that time. Also not used in the normalization of ENDF/B-V were the averages over the ^{252}Cf - spontaneous fission neutron spectrum. ENDF/B-V is ~1.3% higher in normalization than the present evaluated result with which it is compared in Fig. 20. ENDF/B-V values are given in Table. VI.

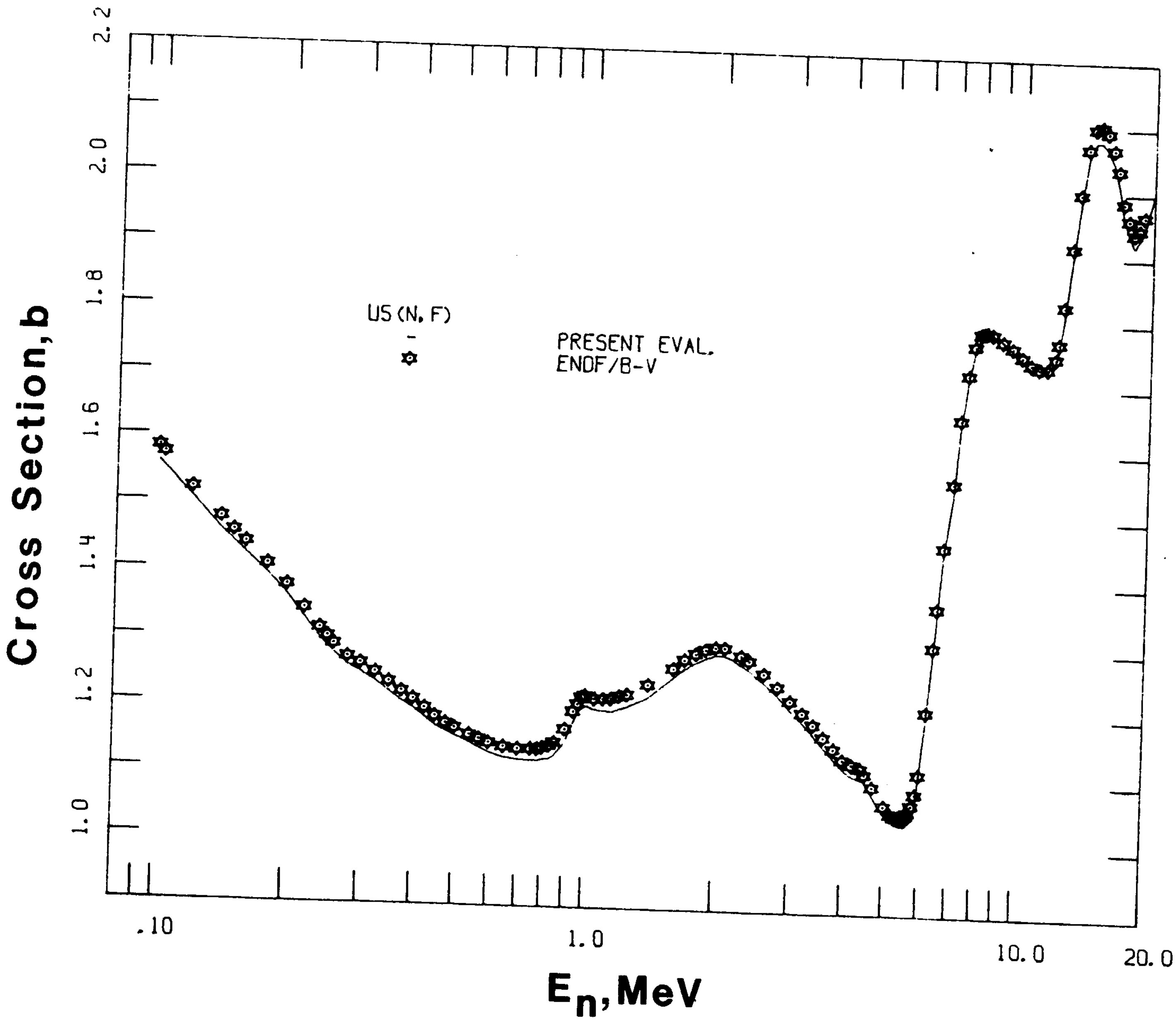

Fig. 20. Comparison of the Present Evaluated $^{235}U(n,f)$ Cross Section with ENDF/B-V.

Table VI. $^{235}U(n,f)$ of ENDF/B-V

E_n, MeV	σ, b	E_n, MeV	σ, b
0.100	1.581	3.400	1.184
0.103	1.572	3.600	1.165
0.120	1.520	3.800	1.148
0.140	1.476	4.000	1.132
0.150	1.457	4.200	1.125
0.160	1.440	4.400	1.120
0.180	1.408	4.500	1.111
0.200	1.377	4.700	1.092
0.220	1.343	5.000	1.064
0.240	1.314	5.200	1.052
0.250	1.302	5.300	1.048
0.260	1.291	5.400	1.047
0.280	1.272	5.500	1.047
0.300	1.262	5.600	1.049
0.325	1.249	5.640	1.051
0.350	1.235	5.700	1.055
0.375	1.221	5.800	1.066
0.400	1.209	5.900	1.083
0.425	1.196	6.000	1.112
0.450	1.184	6.200	1.207
0.475	1.174	6.400	1.306
0.500	1.167	6.500	1.364
0.540	1.157	6.700	1.456
0.570	1.151	7.000	1.553
0.600	1.145	7.250	1.650
0.650	1.140	7.500	1.719
0.700	1.137	7.750	1.763
0.750	1.137	8.000	1.782
0.780	1.137	8.150	1.784
0.800	1.139	8.250	1.784
0.830	1.142	8.500	1.782
0.850	1.147	9.000	1.772
0.900	1.168	9.500	1.762
0.940	1.195	10.000	1.749
0.960	1.207	10.500	1.738
0.980	1.217	11.000	1.732
1.000	1.220	11.500	1.732
1.050	1.215	12.000	1.748
1.100	1.215	12.200	1.771
1.150	1.216	12.500	1.826
1.200	1.220	13.000	1.915
1.250	1.223	13.500	1.998
1.400	1.239	14.000	2.068
1.600	1.264	14.500	2.099
1.700	1.278	15.000	2.103
1.800	1.288	15.500	2.093
1.900	1.294	16.000	2.068
2.000	1.298	16.500	2.036
2.100	1.297	17.000	1.986
2.300	1.286	17.500	1.960
2.400	1.278	18.000	1.939
2.600	1.259	18.500	1.945
2.800	1.240	19.000	1.966
3.000	1.219	19.500	1.990
3.200	1.201	20.000	2.024

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